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ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS: CRITICAL ISSUES AND RECOMMENDED RESEARCH

by

W. J. Shack, T. F. Kassner, D. S. Kupperman,
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ABSTRACT

This report addresses the problem of environmentally assisted cracking in light-water reactor components. The field experience is reviewed, the current utility/industry effort to solve the problem is discussed, and the remaining unresolved technical issues are identified. The additional research and development needed by the NRC to develop an independent capability to evaluate current stress-corrosion cracking problems in light-water reactors and their potential solutions is assessed. The problems and research are divided into six general areas: (1) Leak Detection and Nondestructive Evaluation, (2) Analysis of Sensitization, (3) Analysis of Crack Growth, (4) Nonenvironmental Corrective Actions, (5) Environmental Corrective Actions, and (6) Basic Understanding of Stress-corrosion Cracking. It should be noted that this report was essentially completed in December, 1980. Although additional information and research results continue to appear, the report still serves as a useful summary of the work up until the end of 1980 and outlines the critical areas that are the subject of current, ongoing research.

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A2212-0	Environmentally Assisted Cracking in LWR Piping Systems

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ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS: CRITICAL ISSUES AND RECOMMENDED RESEARCH

EXECUTIVE SUMMARY

Stress-corrosion cracking (SCC) is a generic problem in light water reactors (LWRs). In Boiling Water Reactors (BWRs), the problem has escalated in recent years from cracks in small-diameter (e.g., 4-in.) piping to cracks in the Emergency Core Cooling System and the largest (e.g., 24-in.) primary-system recirculation lines. The cracks, normally located in sensitized stainless steel, are difficult to detect until substantial leakage occurs. A major utility/industry effort is under way to address these problems. To date, most progress has been made in providing additional margin against SCC in new plants through alternate materials and fabrication procedures. For existing plants, the most promising remedy for the prevention of SCC appears to be through modifications of the coolant water chemistry, although only preliminary assessments of this approach have been made.

In Pressurized Water Reactors (PWRs), SCC is usually found outside the primary system, since the oxygen level in the primary coolant is kept low through additions of hydrazine and a hydrogen overpressure. It has occurred recently in PWR secondary systems, including the decay-heat-removal system and spent-fuel-pool piping system. PWR steam generator tubing has also experienced SCC, and most recently, SCC has occurred in PWR turbine discs. Recent cracks in steam generator feedwater piping, however, have been attributed to thermal fatigue.

Detection of SCC through normal ultrasonic in-service inspections is very difficult. As a result, many through-wall cracks have occurred, and in-

creased reliance must be placed on leak-detection systems. Current efforts toward improving in-service inspection are focused on the use of pattern recognition and signal-processing techniques to reduce the dependence on the judgment and experience of the operator. Acoustic emission techniques for leak detection are promising, but require additional research and development.

Although the current understanding of the mechanisms of SCC in LWRs is incomplete, it is generally recognized that three primary factors are required: (1) a susceptible material, (2) a tensile stress, and (3) a corrosive environment. Sufficient modification of these conditions may produce immunity to SCC during the lifetime of an LWR system.

Austenitic stainless steels can become susceptible to intergranular stress-corrosion cracking (IGSCC) as a result of chromium depletion near the grain boundaries by chromium carbide precipitation in grain boundaries of the heat-affected zones next to welds or heat-treated components. Nucleation of the carbides, even without significant growth at the high welding or heat-treating temperatures, may lead to susceptibility with time at reactor operating temperatures. This latter phenomenon is referred to as low-temperature sensitization (LTS). Although the kinetics of LTS are not well understood, it is clear that plastic strain accelerates the process; quantitative measures of sensitization are being developed, as well as improved welding techniques to reduce the degree of sensitization. Alternate materials have been proposed that attempt to avoid sensitization through lower carbon content or through alloy additions that promote the formation of other carbides not containing chromium. The problem of LTS, however, has not been fully explored for either austenitic stainless steels or alternate materials.

Residual stresses resulting from weld shrinkage together with operating stresses appear to provide the driving force for IGSCC, and fabrication techniques that induce compressive stresses on the inner surface of the weldment seem to increase the resistance to IGSCC. In larger-diameter piping there is limited evidence of reduced tensile surface residual stresses and large compressive mid-wall stresses. This may account for the much lower frequency of cracking observed in the larger-diameter lines and for the geometry of the cracks that have occurred.

Several fabrication techniques and alternate materials have been developed for use in new construction or repairs, which may greatly increase the margin against IGSCC. An extensive testing program is being carried out by industry to qualify these techniques and materials. Full-scale piping tests are being used to determine factors of lifetime improvement for the remedies in comparison with Type 304 stainless steel. The quantitative meaning of these factors is open to question, but the qualitative results are encouraging.

For existing BWRs, the addition of hydrogen to the primary feedwater, along with deaeration during operating transients, is currently under study as a means of lowering the oxygen level in the primary system. The role of other ionic species in the coolant water must also be better understood, and the question of potential side effects from chemical modifications must be resolved. Other remedies that can be applied to existing plants are based on changing the state of residual stress on the inner surface of a weldment from tensile to compressive. The effectiveness and feasibility of these techniques have not yet been fully established, although extensive research programs to assess them are being carried out in the US and Japan.

Since it is now clear that IGSCC can occur anywhere in the recirculation system of BWRs, including main recirculation lines, it is important to be able to evaluate the margin against failure remaining in lines with shallow cracks. Experimental data on crack propagation rates and residual stress distributions near weldments are being obtained. Analytical models to predict remaining lifetimes and crack shapes are being developed. However, these models are based on linear elastic fracture mechanics (LEFM), and the applicability of LEFM to the analysis of SCC propagation requires further study.

Despite the large investment by industry in various aapects of the SCC problem, only very limited efforts have been devoted to the development of design approaches through a fundamental understanding of the problem, although some recent work in this regard is promising.

Based on the foregoing considerations and an analysis of current efforts by industry, the additional research and development needed by the NRC to develop an independent capability to evaluate current SCC problems in LWRs and their potential solutions is assessed. Six general areas of research are considered: (1) Leak Detection and Nondestructive Evaluation, (2) Analysis of Sensitization, (3) Analysis of Crack Growth, (4) Nonenvironmental Corrective Actions, (5) Environmental Corrective Actions, and (6) Basic Understanding of SCC in LWRs. Assuming the problems do not escalate substantially in the meantime, it is estimated that a five-year period will be required to complete the effort, lagging the much larger efforts of the utility industry by about two years.

I. INTRODUCTION

Stress-corrosion cracking (SCC) is a generic problem in light water reactors (LWRs) that has occurred in a number of different components. As early as 1965, intergranular stress-corrosion cracking (IGSCC) was observed in Boiling Water Reactor (BWR) piping. In 1974, a number of cracking instances occurred in several BWRs, typically in smaller-diameter (i.e., $\leq$10 in.) lines. Cracking continued to occur in these lines and was also found in BWR feedwater nozzles in 1976. In 1978 cracks were observed for the first time in a large-diameter (24 in.) primary recirculation line. Also in 1978, after a throughwall leak had been detected, all eight of the 10-inch recirculation-inlet-nozzle safe ends on the Duane Arnold BWR were found to have cracks, which extended almost completely around the circumference and penetrated more than half the wall thickness. The severity of the incidents (both in terms of the safety implications and the difficulty and expense of repair) appears to have escalated as the problem has spread. Cracking incidents continue to occur in BWR piping, despite a major effort by the utilities and industry to solve the problem.

Stress-corrosion cracking has also been a concern in Pressurized Water Reactors (PWRs), particularly in recirculating (U-tube) steam generators, since concentrations of chlorides and other impurities from condenser leakage are difficult to avoid. Despite attempts to improve steam generator design and control the water chemistry, crevices and dryout occur, with resultant locally caustic or acidic conditions. In 1975, recirculating PWR steam generators that had switched from a coordinated phosphate water treatment to an all-volatile

treatment (AVT) began to experience tube "denting". In denting, stress is generated on the tubes by the buildup of corrosion products at the tube/support plate intersections. This can then lead to SCC of the tubes. This generic problem has resulted in the forced replacement of an entire PWR steam generator and planned replacement or re-tubing of several others. In 1979, thermal fatigue occurred in a number of PWR steam generator nozzle connections, and IGSCC occurred in borated water lines. Most recently, in 1980, SCC has been postulated as the cause of a serious outbreak of cracking in PWR turbines.

In response to these events, the Nuclear Regulatory Commission (NRC) convened a Pipe Crack Study Group on three different occasions to evaluate the safety implications of the cracking. In addition, the NRC has sponsored several specific research efforts and has issued several regulatory guides, in an attempt to understand the various problems, detect them early, and mitigate their consequences. The utility industry, in turn, has mounted major efforts to develop solutions to these problems, with a five-year, ∿$44 million program on BWR-SCC problems and a five-year, ∿$40 million program on PWR steam generator problems currently under way. These major research programs are sponsored by the owner utilities and are being managed through the Electric Power Research Institute (EPRI). EPRI is also initiating programs on turbine cracking and borated water line cracking. The reactor vendors are actively involved in all these efforts.

It is clear that many suggestions for dealing with the problems of SCC in LWRs will be forthcoming from the efforts of the utilities and industry. In order to develop an independent technical assessment of the problems, and to be better

prepared to evaluate the viability of proposed solutions, the NRC has commissioned the current effort.

The objective, then, is to develop a program addressing the critical SCC technical issues, to arrive at an independent capability to evaluate the problems, and to evaluate the technical merits of proposed solutions. This report develops a program plan to accomplish that objective. After a review of service experience, critical technical issues are identified. Current research efforts addressing these issues are reviewed and the additional research and development needed by the NRC to develop an independent capability to evaluate current SCC problems in LWRs is assessed. A summary of related programs is included in the Appendix.

II. BACKGROUND

Stress-corrosion cracking occurs when a susceptible material is subjected to tensile stress and a corrosive environment. In LWRs, a number of materials, most notably weld-sensitized Type 304 austenitic stainless steel and Inconel 600, have proved to be susceptible to SCC. The high-stress conditions often result from a combination of residual stresses due to welding or other fabrication procedures, various operating stresses (including thermal stresses), stress concentrations, and/or expansion stresses due to the buildup of corrosion deposits in crevices. Environmental influences conducive to SCC include oxygen, peroxide, chlorides, caustic, and other contaminants in the cooling water; crevices or other unfavorable thermal/hydraulic conditions can produce concentrated levels of these species.

When existing LWRs were built, designers attempted to avoid SCC primarily through a judicious choice of materials (necessarily based on the results of short-term tests) and the specification of fabrication procedures believed appropriate at that time. Today designers benefit from the field experience accumulated over the last several years, but their ability to prevent SCC still basically relies on the choice of resistant materials, improved fabrication procedures, and improved thermal/hydraulic design. Attempts to prevent SCC through modifications of the environment or stress state in existing units have met with only limited success to date, although considerable research is expected to continue in this area.

A. BWR SCC Problems

1. Pipe Cracking

The occurrence of IGSCC in LWR systems began as early as 1965, with cracks in furnace-sensitized and heavily cold-worked piping in the Dresden I

Boiling Water Reactor.[1] It was not recognized as a generic problem, however, until late 1974 when a number of cracks occurred in several BWRs. At that time, cracks were found in 4-in.-diameter lines, which are used to bypass recirculation loop valves,[2] and in 10-in.-diameter reactor core spray lines,[3] which are a part of the Emergency Core Cooling System (ECCS).* The cracks all occurred in either furnace-sensitized or weld-sensitized austenitic stainless steel. Except for the cracking in furnace-sensitized safe ends, the cracks were circumferential and located in the heat-affected zone (HAZ) adjacent to pipe weldments. A number of different heats of Type 304 stainless steel were involved. All the cracks had initiated on the inside surface of the pipes; some were throughwall, but most were not. Only about half the cracks were detected through normal in-service inspection procedures. No strong correlation was found between plant operating history and the incidence of IGSCC, although there is some evidence that a correlation might exist with the number of plant shutdowns lasting more than 24 hours.[4] These cracks raised three safety concerns. First, that a crack might propagate rapidly enough to cause a double-ended pipe rupture before detection, thus initiating a loss-of-coolant accident (LOCA); second, that the cracks in the core spray lines might cause partial failure of the ECCS in the event of a LOCA; third, that IGSCC might also occur in non-isolable sections of the large-diameter (e.g., 26 in.) primary recirculation lines.

In response to these concerns, Chairman Anders of the NRC testified to the Joint Committee on Atomic Energy as follows:

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*For details of plants and pipe configurations, see Refs. 2-5.

> In judging the safety implications of the small cracks in the 4 inch bypass and 10 inch core spray lines, several factors were considered. The stainless steels involved are ductile so that the growth of such cracks proceeds slowly and detectable leakage would result well before complete pipe failure. All the plants contain sensitive leakage monitoring systems to detect developing leaks, thus enabling plant shutdown prior to significant pipe failure.
>
> Furthermore, the redundant emergency core cooling system provided on boiling water reactors protects against the consequences of complete pipe failures ranging from the smallest to the largest pipes in the primary system, and it should be realized that in no case were past cracks large enough to produce leakage sufficient to cause any significant drop in coolant levels in the reactors or to involve activation of the emergency core cooling system. These factors reflect the "defense in depth" concept utilized in the design and operation of nuclear power plants.[5]

The Pipe Cracking Study Group, convened by the NRC, followed this with a detailed report issued in October, 1975.[6] Their report concluded that total severance of the ductile austenitic stainless steel piping without prior leakage was extremely unlikely, and in any event, the ECCS and containment were provided to cover such an occurrence. Further, even if leaking cracks were present in the core spray piping, the core spray system should operate as expected. The group added, however, that

> Although the probability is extremely low that the presence of cracks and leaks could result in a significant safety hazard to the public, the presence of such conditions is not consistent with the Design Criteria in 10CFR50.* Steps should be taken to minimize stress corrosion cracking in BWR piping systems to eliminate this condition and also improve plant reliability.[7]

Two subsequent events have occurred, however, that caused the NRC to re-evaluate the safety implications of pipe cracking in BWRs. In 1978, during a

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*US Code of Federal Regulations, Title 10, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants".

plant outage, a crack was found visually on the inner surface of a 24-in.-diameter primary recirculation line in the Gundremmingen ("KRB") German BWR. The crack was very shallow, however, and appeared to terminate at the weld metal of the adjacent weld. This changed the question of whether a crack could occur in such a line to a question of whether such a crack is capable of propagating rapidly. Also in June 1978, it was discovered that all eight of the 10-in.-diameter recirculation-inlet-nozzle Inconel 600 safe ends at the Duane Arnold BWR had cracks which extended essentially completely around the circumference of the pipes and penetrated 50% to 75% of the wall thickness from the inner surface.[8] The cracks were found only after one of them had been leaking for several days, and even when the pipes were subsequently examined by ultrasonics, the nondestructive inspections failed to detect the cracks in two of them. The Duane Arnold situation also raised several safety issues:

(a) In the case of IGSCC, what effective margin of safety can be assumed between the occurrence of a small leak and a major LOCA?

(b) Can a small unidentified leak rate be tolerated during BWR operation?

(c) How strong is the nondestructive evaluation (NDE) line of defense?

(d) Could pipes in this condition survive a major shock loading such as an earthquake or water hammer?

(e) Is the ECCS adequate to handle multiple, non-isolable, through-wall piping cracks occurring in several lines simultaneously?

In view of these concerns, the NRC reconvened its Pipe Crack Study Group. The group reached the following conclusion in its report, issued in February 1979:

> Based on our review, we have concluded that, although through-wall and part-through-wall IGSCC can exist in BWR stainless-steel piping during plant operation, it is unlikely that significant cracks would go undetected. In addition, based on our review and analysis, we also conclude that it is unlikely these cracks will cause unstable crack growth and excessive loss of coolant. Further, since the Emergency Core Cooling System (ECCS) provides protection should a loss-of-coolant accident occur, we conclude that IGSCC in these lines, while generally undesirable, will not be a hazard to public health and safety.[9]

Regarding the Duane Arnold case specifically, they concluded that

> the unfavorable stress and environmental conditions are considered unique to the particular attachment configuration and the associated crevice formed by this configuration. Further, there is no evidence to indicate that Inconel 600 is particularly susceptible to IGSCC in the absence of a crevice.[10]

Very recently, the failure analysis of a cracked 10-in. elbow from the Quad Cities BWR has complicated the IGSCC problem.[11] In this case, although located in an HAZ, the cracks were axial instead of circumferential, and while most of them stopped at the weld, at least two penetrated the weld metal, with one extending completely through the weld to the pipe on the other side. Thus, the capability of weld metal to arrest IGSCC should not be taken for granted.

In view of the continued occurrence of IGSCC, the NRC has issued NUREG report 0313, Rev. 1, requiring more frequent inspections of the target lines, and recommending repair procedures that should result in improved resistance to IGSCC.[12]

2. Feedwater Nozzle Cracking

In 1976, BWRs experienced a series of feedwater nozzle cracks that were generic and of considerable concern although not primarily a stress-corrosion problem. It is believed that the cracks originated in the austenitic stainless steel cladding of the vessel as a result of thermal fluctuations caused by backflow mixing of the colder feedwater with the reactor vessel water during start-up conditions. This backflow occurred between the stainless-steel-clad feedwater nozzles and a slip-fit stainless steel thermal sleeve. The thermal expansion mismatch between the carbon steel nozzle and its stainless steel cladding, and the low thermal conductivity of the cladding, led to small cracks in the cladding. These cracks were then driven into the carbon steel by the pressure and thermal stresses associated with normal reactor operations. The solution adopted was to redesign the feedwater sparger, nozzle, and thermal sleeve, using an Inconel safe end with an Inconel thermal sleeve welded in, and to eliminate the stainless steel cladding over the carbon steel nozzle area.

This problem was important in that it highlighted a mechanism for generating cracks in reactor vessel cladding and showed that such cracks could then propagate into the carbon steel vessel wall. Although SCC was not identified as the primary cause, once the cracks were initiated, the conditions were present for propagation by SCC.

The solution adopted has apparently been successful; however, elimination of stainless steel cladding in the nozzle area exposes the carbon vessel steel to the BWR primary coolant. This is not of concern for general corrosion and crud formation, since the area is small; but it is of potential concern for environmentally enhanced cracking in the vessel. Although BWR service

experience with carbon steel has been very good to date, laboratory tests have indicated that once initiated, under some circumstances a stress-corrosion crack can propagate through carbon steel almost as rapidly as through stainless steel.[13,14]

3. Control Rod Drive System

Cracking has been observed in control rod drive collet housings of BWRs.[15] The apparent cause was exposure to cold, oxygenated water released from the condensate storage tank to cool the control rod drive system during scrams. This water produced thermal stresses and provided an aggressive environment. The housings, which were fabricated from Type 304 stainless steel, had become sensitized during a nitriding operation. The cracks initiated at a region of the outer surface that coincided with a change in wall thickness. A 90% throughwall penetration was observed in one housing, and another housing had a crack that had propagated essentially around the circumference, with 25% throughwall penetration. At least 66 housings out of 109 examined in the Dresden III BWR were cracked. However, the laboratory analysis indicated that there was no immediate likelihood of the housings developing complete separations, and the likelihood of several of them failing simultaneously during a single reactor scram appeared negligible.

The original housings were replaced with the same material, but exposure to the cold oxygenated water was eliminated by using water from the plant hotwell for cooling, and the problem has not recurred.

4. Reactor Water Cleanup Lines

IGSCC has also been found in the HAZ of welds in small-diameter ($\leq$8 in.) stainless steel BWR water cleanup lines. Since these lines are of much less safety concern, however, they will not be discussed further.

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B. PWR SCC Problems

In general, PWRs have experienced far fewer problems with SCC than BWRs, since PWRs use hydrazine additions and a hydrogen overpressure in the reactor vessel which results in very low O_2 levels, and often use centrifugally cast stainless steel piping in their primary systems. The hydrazine additions and hydrogen overpressure maintain the oxygen level of the primary coolant below about 50 ppb, which is below the threshold currently thought to be required for IGSCC in stainless steel. The centrifugally cast stainless steel piping has a duplex microstructure which is resistant to IGSCC (although it is often difficult to inspect by ultrasonic means). Consequently, instances of SCC in PWRs have arisen outside the primary coolant system. Nevertheless, a number of serious pipe cracking instances have occurred in PWRs in recent years, and the NRC convened a PWR Pipe Crack Study Group in 1979 to review these problems.[1]

1. Borated Water Lines

Stainless steel lines in PWR secondary systems have experienced IGSCC since 1974. The lines have usually contained borated (H_3BO_4) water, often at ambient temperature in an air-saturated condition. Cracks have been found in decay heat removal systems and, more recently, in spent fuel pool piping systems. The cracks, as in BWRs, are found in the HAZ next to welds, originating at the inner surface and propagating circumferentially around the pipe. In 1979, IGSCC was found in these systems at the Three Mile Island Unit I PWR. At the request of the NRC,[2] these systems were subsequently inspected in other PWRs, and cracks were found in at least eight other plants. In all cases the cracks follow the same pattern as outlined previously; they occurred in the HAZ next to welds in Type 304 stainless steel, in low-temperature systems with air-saturated (8 ppm O_2) water. The welds in these lines were not fabricated to ASME specifications, since they do not form part of the primary

pressure boundary. As a result, they would be expected to have more stress concentrations and to be more difficult to inspect.

2. Steam Generator Feedwater Piping

In 1979, throughwall cracks were found in steam generator feedwater pipe elbows (A106B carbon steel, 16 in. diam) at the D. C. Cook Unit 2 PWR. Radiographic examination then revealed cracks in all four elbows on Unit 2, and in three of the four elbows on Unit 1. Upon subsequent removal, all eight elbows were found to contain circumferential cracks.[3] The cracks were transgranular, appeared to start from a machined change in section thickness (a counterbore for a weld) on the inner surface, and were initiated by pitting. The cause was identified as corrosion fatigue in an early failure analysis,[17] but a later study concluded that the cause was thermal fatigue.[1]

Subsequently, the NRC requested inspections of all PWR steam generator feedwater systems, and cracks were found in all the elbows (or reducers in some plant designs) adjacent to the feedwater nozzles in eleven different PWRs. The cracks extended around up to 84% of the circumference of the pipes, and over halfway through the wall thickness.[4] These cracks therefore pose safety questions similar to those associated with the Duane Arnold BWR vessel nozzle cracks. These questions were addressed by the NRC PWR Pipe Cracking Study Group.[1]

Further analysis of the steam generator feedwater piping cracks, by a utility/owners group,[5] has demonstrated that stresses resulting from thermal layering under low-flow conditions are the root cause of the cracking. The

thermal stratification in the pipe results when colder water from the auxiliary feedwater supply system meets the hotter steam-generator water that backflows into the pipe. The resultant thermal striping causes fluctuating thermal stresses, and leads to corrosion fatigue. Solutions under consideration for this problem include the use of a thermal sleeve where possible, and/or a preheater for the auxiliary feedwater supply.

3. Steam Generator Internal SCC

Water chemistry control in large steam generators has always been difficult, owing to impurities in the feedwater and concentrating conditions associated with thermal hydraulics (e.g., dryout) and the presence of crevices. In plants where the pH of the secondary water is controlled by chemical additions (e.g., sodium phosphates), an error on the acidic side could lead to generalized corrosion, and an error on the basic side could lead to SCC of the steam generator tubes.

In 1974, PWRs with recirculating steam generators began switching from phosphate chemical control to an AVT similar to that used in once-through steam generators, but without the additional full-flow demineralization that is normally required with once-through units. Soon after this change, a new problem, tube denting, appeared.[6] Tube denting arises when an acidic condition occurs in the crevice between a tube and a support plate, which leads to accelerated corrosion of the support plate. The corrosion produce, magnetite (Fe_3O_4), is less dense than the carbon steel of the original plate, and the resulting expansion stress "dents" the Inconel 600 tube. As this distortion progresses, the tube develops cracks, the support plate can warp and crack, and unless the process is halted, the steam generator will eventually have to be replaced.

One PWR has already been replaced for this reason, and replacements have been ordered for several others.

In recent years, the denting problem has spread to all types of recirculating PWR steam generators, in plants cooled with either fresh water or seawater and with or without condensate demineralizers. For new units, the solution appears to be the use of a different material (usually a ferritic stainless steel) for the support plates, and a different, less-confining design for the tube/support-plate intersections. For existing units, however, no solution has been found to date, although the use of fresh-water cleaning and/or inhibitors appears promising for slowing down the rate of denting.

The NRC already has programs under way on steam generator tube SCC and degradation,[7,8] and for that reason it will not be of prime interest in the present program.

4. Turbine Disc Cracking

In 1969, a shrunk-on turbine disc failed in the Hinkley Point A plant in England, and fragments penetrated the turbine casing. The failure occurred during a routine overspeed test after four years of operation. Subsequent inspections revealed cracks in two other discs; in all cases the cracks were intergranular and originated at the highly stressed keyways. Temper embrittlement of the 3Cr-1/2Mo ferritic steel was considered to be a contributing factor, coupled with the concentrating of NaOH in the keyway regions.[9-13]

More recently, a disc failure occurred in the United States at the Yankee Rowe PWR plant. In this case, the fragments did not penetrate the turbine casing. Examinations have revealed cracks in several other plants; the phenomenon was the subject of a seminar sponsored recently by EPRI.[14]

The general consensus at the seminar was that such cracking must be expected to continue to occur with present plants. Very little is definitively known about its causes or possible remedies. An "immune" material was discussed, but other investigators stated that they had succeeded in cracking the new alloy in laboratory tests. Cracking can be induced in "high-purity" water as well as contaminated water, and temper embrittlement may not be of great significance. There is a higher cracking incidence with higher yield strength, which suggests that heat treatments which decrease yield strength as much as feasible may be beneficial. Cracking apparently begins in a transgranular mode and switches to an intergranular mode with further propagation. This type of SCC was recently reviewed by Lyle et al.[15] A recent study with a 2.25Cr-1Mo steel[16] showed greatly enhanced susceptibility to SCC in a hot ammonium nitrate solution with only fractions of a monolayer of phosphorus segregated to the prior austenite grain boundaries. Lanthanum additions to tie up the P in precipitates produced marked improvement. Fundamental studies are under way at GE under EPRI sponsorship.[18]

Although the costs associated with turbine failures can be extremely high, the principal safety concern is with possible penetration of the turbine casing and the generation of missiles which might then produce further damage to the reactor plant, and studies are under way to assess this aspect of the problem.[14]

Cracking to date has been limited to Westinghouse-designed rotors which have a significantly higher design stress and contain much tighter crevices. Higher oxygen levels may be implicated, since a BWR plant (with Westinghouse rotor) shows the most extensive cracking to date.[19]

In spite of the possible importance of this problem, manpower and budgetary limitations preclude the inclusion of its study in the initial phase of the proposed program. However, it is expected to be addressed later.

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III. SUMMARY OF CURRENT KNOWLEDGE AND APPROACHES

A. Leak Detection and Nondestructive Evaluation

1. Leak Detection

Early detection of leaks in nuclear reactors is necessary in order to detect deteriorating or failed components and minimize the release of radioactive materials. Before a nuclear power facility can be placed into operation, the NRC requires that operational leak-detection systems of various kinds be installed. This NRC position is stated in Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." It is assumed that the equipment cannot be perfectly leak-tight, and some allowance is therefore made for leakage from packing, shaft seals, and other equipment. These leaks are called identified leaks, and their flow is monitored during plant operation. Thus, even with the system operating normally, there may be some accumulation of water in the sumps with an increase in the level of radioactivity. However, it is expected that these levels of radioactivity and leakage will be reasonably constant with time.

In NUREG-0313 (Rev. 1),[1] augmented in-service inspection (ISI) and leak-detection requirements are presented for pressure-boundary piping. The degree of augmented ISI depends on whether or not piping is classified as "service-sensitive" (susceptible to cracking because of a combination of high local stress, material condition, and high oxygen content in relatively stagnant, intermittent-flow, or low-flow coolant).* Examples of "service-sensitive" lines are core spray lines, recirculation lines, and bypass lines. The augmented leak detection requirements for service-sensitive components are given below.

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*In addition, all lines which cracked previously are automatically included.

From NUREG-0313

Nonconforming Lines that are Service Sensitive

Leak Detection: The reactor coolant leakage detection systems should be upgraded to enhance the discovery of unidentified leakage that may include through-wall cracks developed in austenitic stainless steel piping.

(1) The leakage detection system provided should include sufficiently diverse leak detection methods with adequate sensitivity to detect and measure small leaks in a timely manner and to identify the leakage sources within the practical limits. Acceptable leakage detection and monitoring systems are described in Section C, Regulatory Position of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

Particular attention should be given to upgrading and calibrating those leak detection systems that will provide prompt indication of an increase in leakage rate.

(2) Plant shutdown should be initiated for inspection and corrective action when any of the leakage detection systems indicates, within a period of 24 hours or less, an increase in rate of unidentified leakage in excess of 2 gallons per minute or its equivalent, or when the total unidentified leakage attains a rate of 5 gallons per minute or its equivalent, whichever occurs first. For sump level monitoring systems with fixed measurement interval method, the level should be monitored at 4-hour intervals or less.[1]

The purpose of Regulatory Guide 1.45 is to ensure the detection of leaks from the reactor coolant pressure boundary. However, leaks from such sources as the main feedwater system, reactor coolant pump, and accumulators cannot necessarily be distinguished from pressure-boundary leaks. The limitations of the present code for leak detections has been noted by the Pipe Cracking Study Group:

Although early detection of a through-wall leak is desirable, the actual flow rate of water and/or steam through the leak may initially be low. The water will more than likely have to traverse a long and tortuous path to the sump. Thus, water inventory monitoring for a small leak is not practical. However, because of their sensitivity, radiation monitoring devices should detect an unexpected change in the activity of the containment atmosphere and alert operators. ... If the leakage is substantial, the operators should also be getting information from other leakage detection systems, such as an indicated increase in water flow to the sumps, but they will not know for sure where the leak is coming from and most likely will have to enter the containment for a visual inspection. As discussed in Regulatory Guide 1.45, radiation monitoring devices do not measure leakage in terms of water flow; hence, approximate relationships converting these signals to units of water flow should be formulated to assist the operators in interpreting them. ...

... Although the currently installed leakage detection systems reflect the present state of technology, it is recognized that other detection methods may be developed and used. ... In that the ultimate objective of a leak detection system is to protect the reactor coolant system from more significant degradation than a through-wall leak, perhaps devices such as acoustic emission monitors can be developed not only to detect a through-wall leak, but also to detect and locate the onset of cracking before a leak occurs.[2]

No single leak-detection method combines optimal leakage detection sensitivity, leak-locating ability, and leakage measurement accuracy. For example, while quantitative leakage determination is possible with condensate flow monitors, sump monitors, and primary coolant inventory balance, they are not good for locating leaks and are not necessarily sensitive enough to meet code requirements. Consequently, Regulatory Guide 1.45 requires that at least three different detection methods be employed in the reactor. Sump flow and airborne particulate radioactivity monitoring are mandatory. A third method can involve either monitoring of condensate flow rate from air coolers or monitoring

of airborne gaseous activity. While the current methods used for leak detection reflect the state of the art, other techniques may be developed and used.

The American National Standards Institute (ANSI) has prepared a draft standard[3] which reviews several leak-detection methods and indicates their capabilities for detection, location, and measurement. This information is summarized below.

Radiation monitoring has the potential to detect small leaks, but the sensitivity depends on ambient background and concentration of detectable isotopes, among other things. The leakage can be quantified, but a difficult and lengthy effort is required. Generally, radiation monitors are most effective as detection rather than rate-measurement devices. Both air and liquid radiation monitors are used. Containment air is continuously sampled and ducted through detectors used to sense radioactivity in gases, particulates, and solids. Specific leak location is generally not possible with air monitors, but detection sensitivity is good. Coolant leakage is detectable by liquid monitoring, providing the leak rate is sufficiently high.

Temperature sensors can be located near potential leakage sources where the leak can heat the environment enough to be detected. This temperature rise may be correlated with leak rate. Leak location can also be determined, but the adequacy of this technique depends on volume monitored, separation of sensors, magnitude of normal temperature fluctuations, and sensor time constant. Temperature sensors provide only marginal detection, measurement, and location capability, and are therefore generally used in conjunction with other monitors.

Humidity monitoring detects increases in atmospheric vapor content due to a leak. The system is marginally adequate for detection and measurement and poor for determining location.

Moisture-sensitive tape is used for leak location by wrapping it around the pipe insulation; moisture produced by a leak generates an electrical signal, which can be monitored. Since the tape also changes color when moist, the location of the leak can be determined visually. Since the sensors are on the outside of the insulation leaks can be missed if the water runs between the pipe and the inner wall of the insulation.

Drain sumps eventually collect coolant leakage and thus can be used as quantitative flow monitors. The system is good for detection and quantitative measurement, but poor for locating leaks.

When an abnormal leak occurs, the containment humidity and condensation runoff rate will increase. The measurement of runoff provides a quantitative leak-detection (but not location) technique.

Pressure measurements can be carried out with existing plant monitoring devices. However, the method is not sensitive to small leaks and cannot be used for leak location.

Visual observations can be very effective for locating leaks, but detection and measurement are dependent on the frequency of inspections.

Table III.1, from the ANSI report,[3] compares the effectiveness of the various leak-detection schemes. In general, pipe insulation complicates the situation.

Acoustic methods represent the most promising area for improvement in leak detection. They have the best chance of success when specific welds are monitored, and the least when a few probes are used to monitor many welds. The ability to locate leaks and quantify leak rates using acoustic techniques is not well established, especially since virtually no data are available on acoustic signals from field-induced IGSCC.

It is pointed out in NRC report NUREG-0691, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors," that there has not been any definitive work on characterizing leak rate or leak type by acoustic methods. Investigations are recommended which would lead to improved methods for detecting leaks, estimating leak quantity, and determining leak type. Further research suggested includes establishing sensitivity for detecting leaks from cracks under reactor operating conditions and evaluating acoustic sensors for overall effectiveness in leak detection.

Table III.1. Capabilities of Leakage Monitoring Methods Inside Containment (from Ref. 3)

Method	Leakage Detection Sensitivity	Leakage Measurement Accuracy	Leak Location Detection
Air Radioparticulate Activity Monitor[5]	F (P)[1]	F (P)	F (P)
Sump Monitoring[3,5]	G (G)	G (G)	P (P)
Radiogas Activity Monitor	F (P)	F (P)	F (P)
Air Cooler Condensate Flow Monitors[3]	G (G)	F (F)	P (P)
Humidity-Dew Point	F (F)	P (P)	P (P)
Primary Coolant Inventory[2,3] (based on makeup flow integrator)	G (N/A)[4]	G[6] (N/A)	P (N/A)
Tape Moisture Sensors	G (G)	P (P)	G (G)
Temperature	F (F/P)	P (P)	F (F)
Pressure	F (F/P)	P (P)	P (P)
Acoustic	G (G)	P (P)	G (G)
Visual[7]	F (F)	P (P)	G (G)

[1] G-Good, F-Fair, P-Poor for assessing leakage from reactor coolant pressure boundary. Designations in parentheses ( ) are for leakage from any system inside containment.

[2] For PWR during steady-state conditions.

[3] Quantitative determination possible.

[4] Not applicable.

[5] Regulatory Guide 1.45 requirement.

[6] Leakage measurement accuracy is good if determined over a long time period (6-24 h).

[7] Access to containment during normal power operation restricted; therefore, usefulness of this method limited.

Several years ago, Pollock and Wadin[5] attempted to detect propagating cracks by acoustic emission monitoring. It was not then possible to give convincing evidence that growing cracks could be detected. Background noise was a considerable problem. In a more recent Battelle-PNL effort, a digital memory acoustic emission zone monitor has been used to detect IGSCC.[6] The device was shown capable of detecting the growth of IGSCC in a laboratory environment in studies carried out in 1979 at pipe-cracking test facilities operated by the General Electric Company (GE), San Jose, CA. In this system, acoustic emissions are detected with piezoelectric sensors mounted on waveguides (to keep sensors cool) on either side of a weld. Signal time-of-arrival analysis is used to identify the location of the acoustic signals detected. The data are stored in a digital memory. The system has been shown capable of detecting SCC growth in 4-in. Schedule 80 piping. It is possible that leaks would be detectable with a similar system. Whether or not flow rates can be measured quantitatively is still to be established. However, studies with steam generators suggest that quantitative flow measurements can be made using acoustic techniques.[7,8] The procedure is now being developed for in-service monitoring on LMFBRs.

J. R. Smith et al.[9] of the Westinghouse Electric Corp. have carried out experiments to investigate the possibility of employing acoustic monitors for leak detection in PWRs. Field tests and laboratory studies were conducted under simulated reactor conditions. The laboratory tests were designed to establish the sensitivity and detection capabilities of acoustic emission techniques. Steam leaks through circular orifices in pressurized systems were considered. In-plant monitoring (Indian Point, Consolidated Edison) included signal attenuation measurements, background noise measurements, and leak detection tests on both a natural leak (unseated valve) and an artificial leak (pipe section containing a

needle valve). The results suggest that leaks as small as 80 cm^3/min (0.02 gal/min) could be detected (near the source) in a PWR by acoustic emission techniques, and that acoustic emission monitoring of valve and pipe leakage would be sueful. The report indicates that a valve leak could be located by measuring the amplitude decay of the resultant acoustic signal. Whether or not these results are applicable to BWRs and IGSCC is still to be determined. Other studies using acoustic sensors have been carried out. These include (a) work with the primary piping of the Elk River Nuclear Power Plant during a hot hydrostatic test[10] (the burst nature of the leak was seen in this investigation), (b) background noise measurements on a BWR[11] and a PWR[12], and (c) work in Germany on detection and location of leaks by acoustic means.[13]

Estimates of leak rates for reactor leaks detected by Westinghouse using acoustic techniques include 0.2 gal/min at 20-33 feet (depending on background noise)[14], 1 gal/min at 40 ft (cold leg), and 0.5 gal/min at 10 ft (hot leg)[14,15]; the two latter estimates were for a valve leak. It was not clear whether the background noise was typical in these cases. A discussion of acoustic leak detection work in nuclear reactors is also found in NUREG 0691, Appendix F.[4]

Researchers at Battelle-Columbus are carrying out analytical modeling of critical two-phase flow of initially saturated or subcooled liquid through tight cracks. The model predictions are being compared with experimental data from simulated cracks. Acoustic frequency spectra data are also being studied.

Acoustic emission techniques for leak detection in nuclear reactors have recently been reviewed by EPRI.[55] This survey concludes that work is needed to (a) characterize the flow rate as a function of crack configuration and the attenuation of sound waves in reactor piping, (b) establish the mechanisms of acoustic signal generation, and (c) determine sensor reliability. The report also indicates a need to obtain more information on background reactor operating noise and leak noise frequency spectra.

2. Nondestructive Evaluation

In recent years numerous cracks have been discovered in coolant systems of LWRs. Although the main problems have been in BWR primary coolant lines, cracking has occurred in PWR feedwater piping and low-temperature, low-pressure lines near the spent-fuel storage pool. It is desirable to detect any cracks in the primary and secondary coolant boundary and fix them within a reasonable length of time for safety-related as well as practical reasons. The NRC "defense-in-depth" philosophy requires that all possible efforts be considered which may prevent a nuclear accident.

Many of these cracks were missed during ultrasonic ISI and detected only because of leaking, thereby raising doubts concerning the capability of ultrasonic ISI to detect cracks. An NRC study group points out that present ASME code evaluation standards are inadequate for stainless steel and that advanced detection methods may be required to detect tight, highly branched IGSCC. The report recommends that the development of methods for improved crack detection be actively pursued. A more recent study[4] also concludes that ASME code Section V and XI ultrasonic testing (UT) procedures for ferritic weldments are not adequate for detection and evaluation of IGSCC in austenitic SS piping. The recommendation in this study is that UT procedures for IGSCC should be demonstrated on pipes containing IGSCC before use in the field.

The detection of IGSCC before the cracks have grown large enough to cause a leak and the detection, location, and sizing of leaks once they occur are very difficult technical goals to achieve. IGSCC that can be detected by conventional UT under laboratory conditions may be missed during a field examination even by the most skilled operator.

About 95% of US ISI efforts are carried out by private NDE organizations and reactor vendors; in general, utility personnel do not carry out the actual inspections. Variations in performance do exist between organizations. These are related to differences in search units and other equipment; procedures, including radiographic techniques used to distinguish cracks from geometrical effects; levels of experience of personnel; standards; data recording methods; and crack evaluation criteria. The variation in operator skill and procedures for the ultrasonic detection of IGSCC was illustrated in a round-robin carried out by EPRI,[18] which is discussed in more detail later in this section.

The numerous factors which make effective ultrasonic examination of nuclear piping difficult are generally well known. They include (a) weak ultrasonic echos because of the irregular surface, tightness, and variable orientation (circumferential, axial, or skewed) of cracks, (b) the unknown location and variable geometry of the counterbore on the inner surface, (c) the unknown contour of the weld root bead on the inner surface, (d) variations in ultrasonic attenuation between base metal and weld metal and within the HAZ because of the variation in grain structure, (e) weld noise (ultrasonic reflections from columnar grains or echos from unexpected surfaces due to beam deviations resulting from

the anisotropic nature of the weld metal), (f) a difficult environment (high radiation levels, high temperature and humidity, limited access to pipe, etc.), and (g) unsatisfactory calibration standards.

The geometry of the counterbore and the presence of the weld root bead can result in spurious ultrasonic reflections at the pipe inner surface. This is well illustrated by a study on a pipe-to-elbow section of 4-in. bypass piping, removed from service because of numerous ultrasonic indications.[19] Sectioning of the pipe revealed no cracks; the signals, which varied in amplitude, were all from geometrical reflectors. This problem is not confined to small pipes. A section of 26-in. piping can show signals obtained simultaneously from the root bead and land (step). A crack signal originating between the root and land would probably be obscured by the echoes from the geometrical reflectors. It is difficult to inspect the HAZ close to the weld fusion line because of the weld crown, but grinding the weld crown flat can alleviate this problem. The problem associated with spurious signals from geometrical reflectors is further illustrated in Ref. 20. A 22-in.-dia pipe showed 422 separate indications; essentially all of them were subsequently identified as geometric. It is generally reported that at a recordable signal level of 20% of the distance amplitude correction (DAC) curve, as formerly specified in the code, 1000 geometrical reflectors are recorded for each flaw detected.[20] This, of course, is the motivation for the specification in the ASME code of a higher recordable level (50% DAC), which leads to reduced sensitivity for detection of IGSCC.

The different microstructures in the base metal, HAZ, and weld metal cause variations in ultrasonic attenuation. A variation of up to 7 dB in

apparent attenuation between base metal and HAZ for a 2.25-MHz signal has been noted[21] for a piece of 4-in. piping. A further loss of ~10 dB is expected as the signal passes through the weld metal.

Many welds have only single-side access because of the presence of valves and elbows. The high attenuation in the weld metal makes inspection through the weld metal (to examine the HAZ) extremely difficult except possibly for pipes with thin (< 1/4-in.-thick) walls. While 1/2-V-path (straight-path) inspections are normally used, through-weld inspections may require a 3/2-V path (beam reflecting off inner and outer walls of pipe), further reducing the inspectability. The large grain structure can occasionally result in spurious ultrasonic signals, further complicating the signal interpretation.[21] A slightly textured and thus anisotropic base metal can cause the beam angle to vary from its expected value. This texturing can also lead to variations in echo signal strengths. The weld metal itself is highly anisotropic, and thus the phenomenon of beam deviation plays an important role in the propagation of acoustic waves through the weld metal. Propagation of 45° shear waves through the weld metal is difficult because the beam deviation results in a defocusing effect, thus reducing defect sensitivity. For 45° testing, a longitudinal beam would be more appropriate, as in this mode the beam tends to focus. It is, however, much less practical to use longitudinal waves because of the mode-conversion problems and difficulty in generating 45° longitudinal waves. The principle here is that the beam is guided toward (or focused along) directions of propagation with the highest velocity of sound. For waves propagating at 45° to the pipe normal through a V-type butt weld, the longitudinal velocity is

near maximum, while for shear waves (using conventional polarization) the velocity is very low compared to other propagation directions or polarizations. Yoneyama et al.[22] have shown one example in which an angled shear beam was bent so that it was essentially perpendicular to the surface. Beam deviations of ~45° have been predicted and measured experimentally,[23] thus helping to explain the results of Yoneyama et al.

The conditions under which inspectors must work can be very severe. The need for protective clothing and the high temperature, humidity, background radiation, and noise level can severely hamper the efficiency of the operator. Various studies[24] have indicated the variability in effectiveness between operators. Furthermore, no practical means is available to quantify the effectiveness of an inspector. Although in the future automated equipment may reduce this problem, no currently available equipment of this type can do an effective and reliable job.

The primary difficulty with present calibration procedures is that the signal amplitudes from artificial reflectors (side-drilled holes, flat-bottom holes, or notches) do not correlate well with those from IGSCC. The roughness and tightness of IGSCC result in weak and variable ultrasonic echos. In addition, there can be a variability between calibration blocks as well as in the equipment employed. In Ref. 19, for example, differences in UT response were found between Aerojet Nuclear Corp. (ANC) and EPRI calibration blocks used for a study on flaw detection in 10-in. stainless steel piping. An 0.025-in.-deep notch in the EPRI block gave a signal 8 dB higher than that from an 0.030-in.-deep notch in the ANC block. The beam angle was ~37° for the ANC

block and ∿45° for the EPRI block. The 2% difference in velocity noted could not explain the variation in UT response from the notches.

Since IGSCC as experienced in the field can produce very tight cracks, and since these cracks can be circumferential, axial, skewed, or a combination, sizing as well as detection of the cracks is extremely difficult. Transmission losses due to crack tightness, and scattering losses due to crack roughness, branching, and/or orientation, generally result in underestimation of the crack size. These problems are discussed in more detail in Ref. 26.

It should also be noted that, since Section XI of the ASME code does not specifically discuss the inspection of stainless steel (except for a reference to calibration notches and beam angle in supplement 7 of Appendix III), the method actually used is either that described in Section XI for ferritic steel (with no mandatory guidance) or the general UT method described in Section V. In addition, no specific guidance is provided for detection of IGSCC.

The ultrasonic equipment to be used for ISI consists of a single-probe pulser-receiver and search unit with a frequency of 2.25 MHz and beam angle of 45-70 deg. The calibration of the system requires use of several side-drilled holes varying in diameter as a function of block thickness (e.g., 3/16 and 1/4 in. for blocks 2-6 in. thick, plus an additional 1/16 in. for every additional 2 in. of thickness) as well as notches (10% thickness for austenitic welds). The block should have about the same thickness and acoustic properties as the part being examined. The ASME code specifies that the surface of the part being examined be smooth and free of weld splatter and scale. There should be no rises or depressions which would cause the probe to tilt. The rate of travel of the probe is limited to 6 in./sec. The base metal should be examined to a distance from the weld of 1/2 the wall thickness.

No specific procedures are described for mapping reflectors with respect to weld geometry, nor are codes written for specific ultrasonic flaw detection systems or search units. Since ISI is generally carried out using ultrasonic techniques, while radiography may be used for pre-service inspection, baseline or pre-service UT may not have been carried out on any particular component. The code requirements do not describe the implementation of a recording procedure whereby signal amplitude and metal path are recorded automatically, a method which would enhance signal interpretation. The standardization of inspection techniques could increase the reliability of ISI.

The ASME code may not contain adequate criteria for establishing when an indication should be interpreted as coming from a crack-like defect;[27] furthermore, the threshold levels for a recordable indication have varied from 20 to 100% DAC depending on code edition and addendum employed. The difficulty in distinguishing geometrical reflectors from IGSCC leads to procedures which tends to miss small cracks.

The concern for improving ultrasonic ISI is also seen in the following excerpts from NUREG-0313 (Rev. 1)[1] and the Pipe Cracking Study Group Report, NUREG-0531[2] (also see Ref. 28):

From NUREG-0313

Nondestructive Examination (NDE) Requirements

> In some cases, the code examination procedures may not be effective for detecting or evaluating IGSCC and other ultrasonic (UT) procedures or advanced nondestructive examination techniques may be required to detect and evaluate stress corrosion cracking in austenitic stainless steel piping.[1]

From NUREG-0531

> Although special problems are associated with detecting IGSCC in austenitic materials, conventional equipment in common use will provide a signal from most IGSCC that is clear enough to be evaluated. Exceptionally tight, highly-branched cracks are more difficult to detect, and may require the use of improved detection equipment, such as dual transducers, specifically designed for this purpose.
>
> If present Code evaluation standards are used, many cases of IGSCC will not be properly identified, because the amplitude of the detected indications is too low to be considered significant. The main reason that IGSCC is often not identified by ultrasonic examination is that the detected signals, even from large cracks, are often only about 20% to 30% as large as is required to be evaluated according to present Code rules ...
>
> It is the Study Group's opinion that ultrasonic examination can be effective in identifying most IGSCC before leaks occur if the most susceptible welds are examined at frequent intervals, if equipment especially suited to detect IGSCC is used, and if improved evaluation methods are used.
>
> Although the probability of identifying significant IGSCC by ultrasonic examination cannot be accurately quantified, we believe that most IGSCC deeper than about 10% of the wall thickness and several inches in circumferential length will very seldom be missed. Some very tight, highly-branched cracks may be more difficult to identify; however, we believe that these cracks will be detected before they propagate a significant distance through the wall because of their slow rate of growth, ensuring UT reexamination before leakage.[2]

In a report prepared by ANL for EPRI,[21] ultrasonic pulse-echo techniques for detecting IGSCC in 4-, 10-, and 26-in.-dia Type 304 stainless steel Schedule 80 piping were evaluated. The effectiveness of a variety of ultrasonic

transducers was compared. Transducers with varying frequency, size, and beam angle were used. Focused, nonfocused, and single- and dual-angle-beam transducers were also tested. In addition, curved and flat transducer wedges were compared. Artificial reflectors of varying size and orientation were employed in both welded and unwelded pipes to establish flaw-detection reliability and sensitivity. Data from welded pipe sections with laboratory-grown and field-induced intergranular cracks were also presented. Some limited statistical data for establishing the probability of detection of small reflectors were presented for artificial reflectors in 4-in. pipes and small IGSCC in 10-in. pipes. Problems associated with spurious ultrasonic signals resulting from weld geometry were discussed. Results were also presented on the influence of attenuation in both the weld itself and the adjacent HAZ.

The initial results of the ANL study indicated that no single conventional ultrasonic probe will be optimal for the detection of the various types of cracks that might be expected in nuclear reactor piping (longitudinal, transverse, slanted, skewed). If a flaw of a particular size and orientation is sought, then some improvements can be realized over the present detection capability. Detection of IGSCC can be improved by using a single-element, focused transducer, selecting the optimum transducer size, frequency, and beam angle, and reducing the size of the transducer wedge when there is interference from the weld crown. The probability of detecting field-induced IGSCC in a laboratory environment was estimated for a variety of transducers. Although the actual probability of detecting IGSCC in the field obviously could not be obtained, the relative merits of various transducers for flaw detection in

the laboratory could be assessed. The detection probability varied by almost a factor of 2 between the worst (1-MHz) and best (2.25-MHz focused) transducers.

As indicated, the probability of detecting an IGSCC of a particular size has not yet been adequately established. However, in a round-robin conducted by EPRI,[18] an attempt was made to quantify the ability of various ISI teams to detect field-induced IGSCC. Sections of 4- and 10-in. pipe removed from operating plants, including both cracked and uncracked samples, were used. Five teams from industry performed the examinations. In addition to evaluating the flaw detection and analysis capability of each inspection group, EPRI studied the variables of code interpretation, procedures, techniques, standards, equipment, and training. The majority of the ultrasonic inspections found indications (later confirmed by destructive examinations) in essentially all of the flawed pipes, but interpretation varied as to whether further confirmation was needed to establish the importance of the indications. Comparing the geometric location of the source of suspect signals with the previously established probable location of IGSCCs appeared to be one of the most useful methods of analyzing data.[18]

The report concluded that training of personnel specifically for the detection and analysis of IGSCC was the single most important factor in obtaining a high probability of correct inspection results. Techniques, equipment, and standards all influence the inspection process, but to a lesser degree.[18]

Sizing of IGSCC is extremely difficult, particularly with regard to depth. Current techniques for measuring the length of cracks (the 6-dB drop, 20-dB drop, and vanishing echo) yield different results and depend on corrections

made with regard to the spread of the beam; in these methods, the length of the crack is essentially determined from the location at which the peak crack signal has dropped to a prescribed level. Since no good correlation exists between echo signal amplitude and depth, other techniques must be employed. One possibility for depth measurement is to use the diffraction technique described by Silk and Lidington.[30] In this technique, time-of-flight measurements from the scattered wave at the crack tip are used to estimate depths. Agreement with actual depth (fatigue crack) is better than 15%. However, this technique has not yet been demonstrated to be feasible for IGSCC.

Several attempts have been made to optimize the ultrasonic search unit. In one,[21] a focused single-element probe was found advantageous for detection of small (but not necessarily for large) reflectors. It was also less sensitive to variations in flaw slant angle. A dual-beam probe designed for 4-in. pipes was found advantageous only for detection of skewed reflectors.

In contrast to the above, several other results[31,32] suggest that inspection of stainless steel welds can be improved by the use of tandem (pitch-catch) ultrasonic probes. In a study by A. S. Greer and J. T. McElroy,[31] a tandem (dual-element, pitch-catch) 45° shear-wave transducer was designed specifically for detecting IGSCC in reactor piping. The transducer had a center frequency of 1.5 MHz determined from frequency analysis work indicating a "bandpass region" in the 1.2-1.8 MHz range. This transducer produced IGSCC echoes 7-11 dB greater than those of conventional transducers. Signal-to-noise ratios were also improved.

Improved methods for ultrasonic inspection of stainless steel piping, using the 1.5-MHz dual-element unit, are discussed in Ref. 33. In the examination

of a 26-in.-dia pipe from Germany, it was found that at 50% DAC 17 indications would have been noted with the dual probe but only one with a conventional 2.25 MHz probe. However, the real advantage of the dual probe cannot be established until destructive examinations are carried out.

The results of a GE study using a 1.5-MHz dual probe were also reported in Ref. 18. They indicate that the dual probe offers advantages over conventional search units on large-diameter, heavy-wall piping and on piping with unusual outside geometries.

It is still not entirely clear whether other improvements, such as the use of focused single-element probes and better calibration procedures, would result in inspections as reliable as those obtained with a dual-element probe, since the dual probe does not necessarily lead to better discrimination between cracks and geometrical reflectors. The ability to discriminate signal sources may be more related to calibration procedures and selection of the threshold level for indicating a crack than to probe type.

Since transducer characteristics can have a great effect on reliability of crack detection, it is important to check probes for characteristics such as center frequency, waveform, sensitivity, and beam profile. However, there is variability in what an ISI team will actually do to check the probes. The checks may range from assuming the probe certification from the manufacturer is correct to measurement of center frequency, bandwidth, and beam profile followed by periodic checks on transducer performance. Some groups carry out a DAC calibration and assume the transducer is acceptable if the generated DAC curve is within a few dB of what is typical.

As previously indicated, the lack of ability to distinguish cracks from geometrical or microstructural reflectors is one of the major problems for ISI. Although it is difficult and very time-consuming, most geometrical reflectors ca be distinguished from cracks by proper noting of transducer position relative to weld geometry for all recordable ultrasonic indications. With this information, one can determine whether the reflector is in or out of the HAZ. However, the weld geometry is not always known accurately, and crack signals can still be swamped by signals from geometrical reflectors in the same vicinity as the crack. It is possible that advanced pattern recognition techniques[34] may be useful in distinguishing cracks from geometrical signals, since the scattering pattern and thus the frequency spectrum for cracks (with irregular surfaces) and geometrical reflectors should be markedly different.

The capability to distinguish between geometrical reflectors and cracks in a conventional manual scan will depend upon the skill of the inspectors. Additional effort is required to correlate signals from actual and artificial flaws, and devise other means to determine the size of flaws. An archive of pipes with flaws characterized by dye-enhanced neutron radiography and other methods should be accumulated and made readily available for comparing UT equipment and procedures.

The ability to distinguish geometrical reflectors from cracks may also be improved by more careful mapping of the locations of recordable signals, mapping of the inner surface using normal-incidence longitudinal waves, and the application of radiography to help identify the location of the counterbore and weld root.

To help overcome the shortcomings of existing methods, a number of advanced ultrasonic NDE techniques are being developed. Work on detection of flaws in stainless steel reactor components by adaptive learning networks (ALNs) or pattern recognition techniques is being carried out by Hanford Engineering Development Laboratory (HEDL),[35] Adaptronics Inc.,[34,36] and Zeger-Abrams Inc.[6]

The effort by Adaptronics, sponsored by EPRI, is specifically directed toward detection of IGSCC. The major purpose of this Adaptronics program has been to develop and evaluate a nonlinear ALN signal-processing system to discriminate between geometrical reflectors and flaws during ultrasonic inspection of austenitic stainless steel pipe welds.[34] The first step in developing such a signal-processing system is the selection of a set of signal parameters (such as signal amplitude and power-density spectrum) that can be calculated at each position of an ultrasonic scan. Next, the data (i.e., signal parameters from a collection of flawed components) are divided into three sets -- a training set, a testing set, and a verification set. The training set is used to select signal parameters which appear to characterize flawed components. A signal-processing system or network is developed which incorporates the chosen parameters. The testing set is used to refine and modify the system, and the verification set is used to determine the validity of the final result. The final step is the construction of a field-usable "black box" to actually carry out the signal processing during an inspection. To ensure reproducibility of data, an electronic processing system is generally used with an automatic scanning system to record the circumferential and axial position of the transducer relative to the weld. An example of this is a search unit tracking and recording system (SUTAR) which automatically records ultrasonic signal waveforms and probe position during manual pipe inspections. The system is faster than conventional manual inspections.[25]

Results obtained by HEDL[35] suggest that ALN can produce much greater signal-to-noise ratios than conventional techniques for detection of artificially introduced defects (EDM notches), and the work done by Adaptronics[34,36] indicates that IGSCC can be separated from geometrical reflectors more successfully by ALN techniques than by conventional UT. While these results are encouraging, the applicability of the ALN technique has yet to be demonstrated (to our knowledge) in situations where knowledge of the geometry of the weld and presence or absence of cracks is not available before the test is begun. In other words, the reliability and effectiveness of the method in detecting IGSCC cracks in reactor piping under field conditions has to be established.

In the Zeger-Abrams work[37] (in collaboration with Ultrasonics International), a relatively simple signal-processing system called a "call confirmer" is used to aid manual ultrasonic inspections. The "call confirmer" measures echo signal envelope rise time, pulse envelope duration time, and percentage of power in each of two windows in the frequency spectrum. These measurements are combined to form a measure of whether the signal is a crack or a geometrical reflector, or whether the data are inconclusive. Initial testing indicates that under laboratory conditions the device can distinguish cracks from geometrical reflectors in 4- and 26-in. pipes 90% of the time.[6] However, successful use of this system under field conditions has not yet been demonstrated.

Another approach to ultrasonic inspection that holds considerable promise is the synthetic aperture focusing technique (SAFT) under development at the University of Michigan and Southwest Research Institute. Synthetic aperture imaging takes advantage of the nonlinear phase shift of a target response as it is being linearly scanned. Use of this phase shift allows a more precise determin-

ation of the lateral position of the target, thus resulting in a finer lateral resolution than can be obtained by using the physical beam width alone. Individual A-scans are shifted forward in time by an amount equal to their predicted phase delays; summing them produces an enhanced response from the assumed point target.[37] Flaws have been imaged through six inches of steel by this technique, and the resolution is on the order of one wavelength in both the longitudinal and the lateral direction.[37]

It appears that this technique may provide an effective means of improving spatial resolution and thus flaw characterization, but it does not necessarily provide a means for separating geometrical reflectors from IGSCC signals. In addition, a means for obtaining the information rapidly will have to be fully developed before this technique will be viable for field testing. Efforts to develop rapid flaw detection techniques are being carried out.

Another NDE technique that has potential application to the improved detection of IGSCC is spatial averaging. The problem in inspecting stainless steel, particularly around welds, often reduces to one of extracting the flaw signal from the noise signals. This problem is often related less to electronic noise than to coherent noise, scattering from grain boundaries, rough surfaces or weld-metal interfaces. A signal-averaging method could possibly be used to improve detection of cracks in the HAZ and increase the effectiveness of through-the-weld inspections by averaging out weld-metal noise.

Reference 38 describes work on signal-averaging techniques. Here, A-scan traces are averaged by moving a transducer slightly, thus changing the noise pattern but maintaining a flaw signal. The traces are digitized with a transient recorder and averaged with a minicomputer. Substantial improvements in signal-to-noise rations are seen for detection of fatigue cracks in stainless steel weld metal in flat specimens. Adaptronics is also working with spatial

averaging techniques.[6] In their method, which is essentially SAFT as specifically applied to pipes, pulse-echo waveforms are recorded at different transducer positions around the pipe. The A-scans are delayed in time so the various signals, on the average, add coherently. This technique, called beam focusing by Adaptronics, has been demonstrated to reduce coherent noise significantly for detection of saw cuts in a calibration nozzle. Linzer et al.[29] have also demonstrated the effectiveness of spatial signal averaging in coarse-grain stainless steel.

In conventional shear-wave angle-beam testing of pipes, a longitudinal wave is mode converted at the metal interface to produce shear waves. These waves have a polarization, designated SV, such that the particle motion is in the plane formed by the "full V" path of the acoustic beam. In principle, 45° shear waves with different polarizations could be generated by rotating a normal-incidence shear wave placed on a 45° stainless steel wedge. The use of SH waves, for which the particle motion is perpendicular to the plane formed by the "full V" path of the acoustic beam, could have significant advantages. The SV (conventional) waves can mode convert to longitudinal waves if the angle of incidence is less than about 35°. This can happen if the geometry of the pipe is not ideal or if there are reflections from the weld root. These mode-converted signals could seriously affect the interpretation of the ultrasonic echos. However, the SH waves do not mode convert when reflecting off the inner surface of the pipe. Furthermore, since the weld metal itself is highly anisotropic with respect to velocity and attenuation, the attenuation of shear waves in the weld metal is highly dependent on the shear-wave polarization, and horizontally polarized shear waves (45° to the long axis of the grains) can penetrate the weld metal better than SV waves. Thus, whereas through-weld inspection of the HAZ is impossible with normal shear waves (SV), it may be

possible with SH waves. Beam spreading and beam steering (guiding of the acoustic beam along directions in the weld metal that have the highest velocity of sound) vary with polarization, and defects missed with SV waves may be detected with SH waves. False indications can result from beam steering. The difficulty in using SH waves is that a highly viscous or solid couplant must be used, which makes pipe inspection very slow; alternatively, electromagnetic acoustic transducers (EMATs) can be used. EMAT devices sensitive enough for detection of IGSCC in stainless steel are not yet available. Testing or monitoring with SH waves seems possible now only for systems that can be fixed permanently to the pipe. This would be practical only in the most sensitive areas, or where continuous monitoring is necessary.

Another area of research is in the use of linear arrays for pulse-echo flaw detection and imaging. Linear arrays with up to 240 elements have been designed and evaluated with regard to phase-steered and non-steered applications.[39-40] Although these systems are most appropriate for medical applications, they may offer advantages in special situations involving welds. The main advantage is in the ability to steer an ultrasonic beam without moving the probe. The main problems are in minimizing crosstalk influence and compensating for variations in beam intensity with angle of steered beam.

An important need for the testing and qualification of improved inspection techniques is an archive of flaws that duplicate the characteristics of in-service IGSCC. Efforts have been under way to generate IGSCC in a laboratory environment. Battelle-PNL has an EPRI project to grow and monitor IGSCC under conditions close to field conditions.[6]

The cracks are produced by the crevice technique,[6] using an autoclave for pipes of 10-in. dia or less and a pipe test rig which simulates BWR operating conditions for 26-in. pipes. Ishikawajima-Harima Heavy Industries Co., Ltd. is also conducting an EPRI-sponsored program[6] to demonstrate the feasibility of producing IGSCC of controlled size and location in large pipes. The artificial-crevice technique is also used in this effort. These efforts should provide nonradioactive specimens with tight, non-throughwall IGSCC for use as training samples and calibration standards and in evaluating NDE systems for crack detection, location, and sizing.

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B. Analysis of Sensitization

Austenitic stainless steels such as Types 304 and 316 can become susceptible to intergranular corrosion (IGC) through exposure to temperatures in the range of ~800-1600°F (420-870°C), which causes precipitation of chromium-rich carbides at grain boundaries. The carbide precipitation may decrease the chromium concentration in regions adjacent to grain boundaries to a level below that needed for corrosion protection by passivation. The depleted grain-boundary regions then become susceptible to corrosive attack in certain environments. Such microstructural changes resulting from thermal exposure, which produce IGC susceptibility of stainless steel, are collectively known as sensitization. Sensitization can occur during fabrication processes such as welding, stress-relief heat treatments, and slow cooling through the temperature range noted above. In addition to austenitic stainless steels, a wide variety of chromium-nickel-iron alloys are susceptible to this process. Sensitization can be eliminated by solution heat treatment, or controlled by reducing heat input during fabrication or modifying alloy composition and structure.

Comprehensive reviews on sensitization may be found in the articles by Cowan and Tedmon[1] and Henthorne.[2] The extent or degree of sensitization (DOS) and the rate of sensitization depend on the thermomechanical history, alloy composition, and metallurgical structure of the material.

Diffusion-controlled reaction models for chromium depletion based on the assumption of local thermodynamic equilibrium between chromium and carbon in metallic solution and carbide ($Cr_{23}C_6$) precipitates were developed by Stawström and Hillert.[3] The kinetics of carbide precipitation, the

formation and development of the chromium-depleted zone, desensitization by overaging, the calculation of time-temperature-sensitization (TTS) diagrams, and the effect of nickel concentration were studied, but the effect of strain was not investigated. With 12% or 13% chromium taken as the minimum quantity necessary to avoid sensitization, good agreement was found between experimental IGC data and the prediction of the models. Direct experimental evidence for the chromium depletion model is available. Using a scanning transmission electron microscope with x-ray analyzer, Pande et al.[4] found a Cr-depleted zone 100-150 nm wide in sensitized Type 304 stainless steel. The Cr concentrations were 19% in the bulk grains, 14% at the grain boundaries of sensitized specimens (600°C for 25 h), and 16% at the grain boundaries of stabilized specimens (800°C for 48 h). (It should be noted that the grain-boundary values represent averages of chromium concentration gradients that vary fairly rapidly; the actual minimum concentrations are probably somewhat lower.) Other authors have also found Cr-depleted zones in sensitized Type 304 stainless steel using scanning transmission microscopy,[5] and evidence of the healing of Cr-depletion in overaged material (700°C for 1000 h).[6]

Recently, it has been shown that sensitization can occur at temperatures well below those normally required, if carbide nuclei are present at grain boundaries.[7,8] This phenomenon is referred to as low-temperature sensitization (LTS) and is potentially very important, since it indicates that the degree of sensitization in LWR pipe weldments could increase with reactor operating time, implying that weldments may become increasingly susceptible to IGSCC. The kinetics of the process are slow, but extrapolation of laboratory results

suggests that Type 304 stainless steel pipe weldments may become susceptible in 10 to 20 years at BWR operating temperatures.[7]

Povich[7] and Povich and Rao[8] have demonstrated the existence of LTS by enhanced IGC of Type 304 stainless steel in a modified Strauss solution, enhanced susceptibility to IGSCC by constant extension rate (CERT) tests, and growth of grain-boundary carbides by transmission electron microscopy. Their studies also suggest that, at least at temperatures ranging from 350 to 500°C, LTS is controlled by a rate-limiting process consistent with a diffusion-controlled chromium-depletion model, which obeys an Arrhenius-type temperature dependence with a range of apparent activation energies of 40 to 65 kcal/mol depending on the heat of material. Fullman[9] has reviewed the LTS process, using the chromium depletion model proposed by Stawström and Hillert[3], in order to assess the validity of extrapolation using an Arrhenius-type plot to predict long-time LTS. He concluded that the temperature dependence of LTS in Type 304 stainless steel was consistent with chromium depletion via dislocation-enhanced diffusion, and that Arrhenius extrapolation would provide a reasonable estimate of LTS kinetics. However, reduction of dislocation-enhanced diffusion due to impurity absorption may lower the LTS rate. Hence, the actual rate of LTS at reactor operating temperatures is still unknown. Since LTS-enhanced susceptibility of stainless steel to IGSCC has been experimentally established, evaluation of LWR materials and fabrication processes for possible LTS within reactor lifetimes is important and more work is needed.

Extensive information is available on the effect of isothermal heat treatment on sensitization. This type of data is usually summarized in the

so-called TTS diagram. However, care must be exercised when these data are used to predict the DOS associated with a different type of thermal history, such as continuous cooling.[10] Predictions of continuous-cooling sensitization (CCS) based on isothermal data tend to underestimate the actual degree of CCS. The effects of multiple heating and cooling cycles are not additive, since for sufficiently high peak temperatures, dissolution of the carbides can occur and reduce sensitization. Isothermal TTS curves are used in Ref. 11 to predict the critical cooling rate for sensitization of Type 304 stainless steel, but this approach is not suitable for multiple heating and cooling cycles when maximum temperatures exceed the nose of the TTS curve.

Prior cold work has an important effect on the sensitization of Type 304 stainless steel,[12,13] and in some cases can enhance sensitization.[12] This effect is more pronounced for higher carbon contents and for a lower temperature range of sensitization. It is believed that the cold work increases the dislocation density of the material, thereby enhancing dislocation pipe diffusion of chromium and hence the rate of sensitization. For prior cold work beyond a critical amount, the DOS for a given thermal treatment is reduced below even that of undeformed material, since carbide precipitation occurs rapidly throughout the grain matrix, and no significant carbide precipitation occurs at the grain boundaries. The critical amount of prior cold work depends on alloy composition and temperature.[12] Recently, the effects of straining during heating and cooling cycles of Type 304 stainless steel were investigated.[13] Either tensile or compressive strain on the order of 3-5% or more enhances sensitization, and hence the strain cycles produced by weld solidification shrinkage will result in a higher degree of sensitization

in the HAZ of a weldment than would be expected on the basis of temperature history only. Thus, the results of a furnace sensitization treatment without straining will tend to give an underestimate of the susceptibility of the material to sensitization.

The effects of alloy composition on sensitization are related primarily to the solubility and precipitation rate of chromium-rich carbides. The carbon content is the most significant alloying element affecting the degree of sensitization. Since the solubility of carbon in austenitic stainless steel decreases with decreasing temperature, it is thermodynamically favorable for carbon to precipitate as a carbide. The higher the carbon content, the greater is the tendency to form precipitates. Low-carbon stainless steels, therefore, are promising candidates as alternate alloys in BWRs. General Electric recommends low-carbon (<0.02%) steels with relatively high nitrogen levels (0.06-0.10%); these steels are comparable to conventional austenitic stainless steels in mechanical strength, allowing one-to-one replacement without redesign or analysis.

The effect of nitrogen on sensitization depends on the amount present in the alloy. Some studies[14] indicate that maximum grain-boundary attack in 18% Cr stainless steels occurs for nitrogen levels in the range 0.03 to 0.06%. However, in Type 304L stainless steel containing less than 0.03% carbon, nitrogen additions in the range of 0.06 to 0.15% appear to have no detrimental effect on sensitization and no apparent effect on weldability.[15,16] In fact, some studies indicate that nitrogen additions up to 0.16% retard the sensitization rate.[15]

Increasing the Mo and Cr levels of austenitic stainless steels slightly decreases the susceptibility to sensitization. Full-scale pipe tests[17] show

that Type 316 stainless steel, which contains Mo but is otherwise similar to Type 304 stainless steel, is more resistant than the latter. Additions of Mo to stainless steel also increase resistance to pitting-type corrosion.

Titanium and niobium, which are strong carbide formers, also decrease susceptibility to sensitization. If present in adequate amounts, they can prevent sensitization. No instances of IGSCC have been reported in reactors using stabilized austenitic stainless steel (the stabilized stainless steels Types 321 and 347 contain Ti and Nb, respectively). In contrast, increasing the nickel content of stainless steel increases the susceptibility to sensitization by increasing the carbon activity.[18]

Contradictory results are reported for some alloy additions. Silicon has been shown to decrease resistance to intergranular corrosion,[19,20] especially in Mo-containing steels.[21] However, high levels of silicon (>2.5%) help nonsensitized stainless steels resist pitting and intergranular attack in nitric acid solutions with hexavalent chromium ion. Boron additions have been observed to retard precipitation of chromium carbides in Type 304 stainless steel.[22] However, a deleterious effect of boron was also reported,[2] and further studies are needed to clarify its effect.

Theoretical studies of the effect of alloy composition on sensitization have been reported by Stawström and Hillert,[3] Tedmon et al.,[18] and Fullman.[23] These are based on thermodynamic calculations of equilibrium chromium concentrations adjacent to carbide precipitates. The effects of alloying elements are evaluated in terms of "chromium-equivalency parameters". An element chromium-equivalency parameter is the percent chromium reduction (or addition, when the parameter is a negative number) permitted for a 1% addition of the

element while retaining equivalent DOS. The results of the thermodynamic calculation are in good agreement with experimental results.[24] Overall, the effects of major alloying elements are relatively well understood, but further studies are needed for minor elements such as nitrogen, boron, sulfur, and phosphorus. In particular, the effects on LTS need to be investigated.

There is some evidence which suggests that increasing grain size increases susceptibility to sensitization.[24] This seems reasonable, since the grain-boundary area/carbide-precipitate volume ratio is small for large-grained material, and therefore there is more chance for a network of carbides to form at grain boundaries, but there are not enough data to establish the grain-size effect quantitatively. Recent thermodynamic calculations[23] of chromium concentration adjacent to a carbide precipitate including the effect of the grain-boundary area/carbide volume ratio are in fair agreement with the empirical formulation of Cihal.[24]

Variations in alloy composition, tensile strength, metallurgical structures such as grain size, and the extent of cold work have been noted in piping. Variations in mill annealing practices and other prior thermomechanical treatments which may not be detected by examination of the finished product may also exist. These variations can influence susceptibility to sensitization and IGSCC, and significant heat-to-heat variabilities have been reported in sensitization response and IGSCC susceptibility.[25] However, heat-to-heat variations in sensitization have also been observed, even among heats of Type 304 stainless steel which have similar compositions and microstructure. The reason for this variability among similar materials is not well understood at present, but it may be due to differences in grain-boundary composition and structure which can occur in materials with similar bulk chemistries and structures.

Grain-boundary composition and structure are obviously important for sensitization and IGSCC. Chromium depletion near grain boundaries probably constitutes the best measure of sensitization; however, the segregation of minor alloying elements and the structure of grain boundaries may affect the kinetics and extent of sensitization. The segregation of minor alloying elements to grain boundaries may play a role in heat-to-heat variability in sensitization. It is interesting to note that less susceptibility to sensitization has been observed in commercial alloys than in laboratory-prepared "pure" alloys.[26]

Grain-boundary segregation of minor alloying elements has been investigated using Auger electron spectroscopy.[6,26,27] Enrichment at the grain boundaries of P, S, Si, and N was detected in sensitized Type 304 stainless steel. Segregation of B to grain boundaries was also found in Type 304 stainless steel by Auger electron spectroscopy[26] and in Type 316 stainless steel by optical-microscope autoradiography.[28] Segregation of these elements appears to occur during high-temperature austenization treatments,[26,28] and therefore, it is possible that these elements affect the nucleation and early-stage growth of carbides.

The study of grain-boundary segregation in austenitic stainless steels has been hampered by experimental difficulties. The width of Cr depletion or grain-boundary segregation is typically of the order of 100 nm or less, which means that transmission electron microscopy with very high-resolution analytic capability is needed. It is difficult to fracture these materials intergranularly without altering grain-boundary composition, which limits the usefulness of techniques like Auger electron spectroscopy.

The electrochemical behavior of sensitized grain boundaries of Type 304 stainless steel with and without S segregation has been studied by anodic polarization in sulfuric acid.[27] The primary activation current was larger for segregated than for unsegregated boundaries. The intergranular corrosion of Type 304 stainless steel doped with P, S, Si, or N was studied[26] using Strauss and Huey tests. The results indicated that sensitization is not simply controlled by the chromium and carbon content of material, but is also influenced by segregation of the doping elements. Chromium depletion occurred more rapidly in high-purity alloys than in commercial heats.

Austenitic-ferritic duplex stainless steels are known to be resistant to IGC associated with precipitation of chromium-rich carbides. Detailed studies[29] of the sensitization of Type 308 stainless steel, a duplex stainless steel, have been made. Specimens from 4 wrought alloys and 6 weld-deposited alloys were aged at temperatures ranging from 480° to 700°C for 1/4 to 1000 hours and then tested for IGC by the ASTM 262-E method. For a given carbon content, a critical amount of austenite-ferrite boundary area above which the alloy is immune and below which it is susceptible was proposed. These results are consistent with failure analyses of weldments from operating reactors, which indicate that IGSCC do not normally propagate through Type 308 weld metal. Recently, IGSCC was discovered in weld metal of a 10-in.-diameter Type 304 SS pipe weldment that has a slightly smaller value of austenite-ferrite boundary area (2250 cm^{-1}) than the proposed critical value (2900 cm^{-1}).[30]

Inconel 600, Incoloy 800, and related nickel-base austenitic alloys can also be sensitized by chromium-rich carbide precipitation and become susceptible to IGC. However, some thermal treatments that result in grain-boundary precipitation in Inconel 600 improve IGSCC resistance in deaerated high-temperature water.[31] The effect of sensitization heat treatments on IGSCC susceptibility for these alloys is not yet clearly understood. Thermodynamic calculations on chromium-carbon-carbide equilibria in nickel-base alloys are in fair agreement

with intergranular corrosion measurements.[32] A crevice condition, in addition to stress and sensitization of material, was speculated to have played a role in the intergranular cracks that have been discovered in Inconel 600 safe-ends of the Duane Arnold BWR.[33]

Sensitization may be detected and measured by assessing susceptibility to IGC. Various chemical and electrochemical tests of intergranular corrosion have been discussed in the literature (e.g., Ref. 1). Five standard tests for detecting susceptibility of stainless steels to IGC, designated ASTM A262 Practices A through E,[34] have been adopted by the ASTM. Because of their greater sensitivity in detecting chromium-rich carbide precipitates and chromium depletion and the simplicity of the test procedures, Practice A and E are the most widely used for detecting sensitization in Types 304 and 316 stainless steel.

In Practice A, the specimen is metallographically polished to a final finish of 3/0 or finer, and then etched in a 10 wt % aqueous solution of oxalic acid for 1.5 minutes at a current density of 1 A/cm^2. The etched surface is then examined at a magnification of 250X or 500X, and the degree of sensitization is evaluated by the severity of etching at grain boundaries. In order to quantify the test results, the ratio of the length of ditched grain boundary to the total length of grain boundary has been used as a measure of sensitization.[35,36] Practice A is frequently used as a rapid screening test for accepting materials that are free from susceptibility to sensitization due to chromium-rich carbide precipitation.[34] The oxalic acid etching identifies the presence of such carbides but does not necessarily indicate a chromium-depleted zone since, depending on the thermomechanical history, desensitization may have occurred. There are qualitative correlations between the DOS measured by Practice A and susceptibility to IGSCC, but no quantitative correlations have been established.

Practice E is a 24-hour intergranular corrosion test in a boiling aqueous solution of 16 w/o H_2SO_4 - 6 w/o $CuSO_4$ to which metallic copper shot or grindings are added. It has been widely used in sensitization studies of stainless steel. Following exposure, the specimen is bent 180° and the surface is examined at a magnification of 5 to 20X. The appearance of fissures or cracks constitutes a positive result, and the specimen is considered to be sensitized. In order to quantify the test results, the depth of the intergranular penetration is measured by metallography, or the reduction in tensile strength or electrical conductivity of the specimen is evaluated. Intergranular penetration in the Practice E test occurs in chromium-depleted regions of material.[37] However, the Practice E test is not considered suitable for evaluating IGSCC susceptibility of LWR piping materials, because the sensitivity of the test is limited at low degrees of sensitization.

Although not yet established as an ASTM practice, electrochemical tests[38] for sensitization appear promising. They seem to be much more sensitive at low DOS. The most widely studied test of this type is the electrochemical potentiokinetic reactivation (EPR) technique. Potentiokinetic curves are obtained by a controlled potential sweep from the passive to the active region in an aqueous solution of 1N H_2SO_4 + 0.01N KSCN at 30°C. Activation charge density is used as a measure of DOS. Sensitized steels are easily activated and show high activation charge density. There is good qualitative agreement between the results obtained by the EPR technique and those obtained by ASTM A262 Practice E.[39] Clarke et al.[40] have investigated sensitization of pipe weldments and furnace heat treated specimens from a number of different heats of Types 304 and 304L stainless steel. The EPR results were analyzed for

correlation with the susceptibility to IGSCC in BWR environments as measured by CERT, constant load and pipe tests, and they were also compared with those obtained using ASTM A262 Practices A and E. The EPR technique was found to be superior to the other test methods, since it had greater sensitivity at low DOS and correlated well with the IGSCC susceptibility tests. An acceptance DOS limit of 2 C/cm^2 after welding was proposed for Type 304 and 304L stainless steels.[41]

The EPR technique can also be applied to Type 316 stainless steel. However, the test parameters have to be adjusted to obtain comparable sensitivity.[41] The results for Type 316 and 316L stainless steels are similar to those obtained for Type 304 stainless steel, and the acceptance DOS limit of 2 C/cm^2 proposed for Type 304 and 304L stainless steels also appears to be reasonable for Type 316 and 316L stainless steels.

Efforts have been made to develop a nondestructive EPR field test technique,[42] and portable electrochemical cells for field EPR tests have been developed.[43,44] However, direct measurement of DOS on the inner surface of pipe weldments is not possible in most cases, and it is difficult to correlate with measurements made on the outer surface.

Improvement of the performance of the EPR test by modifying test parameters or procedure may be possible. Recent work[45] indicates that an EPR test performed in an aqueous solution of 1N $HClO_4$ + 0.2N NaCl is more sensitive for detecting the effect of prior cold work on DOS in Type 304 stainless steel than the test in the 1N H_2SO_4 + 0.01N KSCN solution.

The EPR technique, in general, has been shown to be a useful technique for detecting DOS in Type 304 and 316 stainless steels including low carbon grade materials. The applicability of the EPR technique to other alloys (Type 308

stainless steel, Inconel 600, etc.) has yet to be examined. It appears to be satisfactory for qualifying incoming materials for use in construction and thermomechanical treatments during fabrication. The results of a round-robin test among GE and three Japanese laboratories showed good inter-laboratory agreement.[44] A round-robin test among U.S. laboratories has been initiated by the ASTM G-1 Committee, but not yet completed. Anomalous EPR results were noted for a heat of Type 304 stainless steel which had a large grain size.[46] There are also difficulties in applying the EPR technique to sensitized material with prior cold work, since carbide precipitation in grain matrices produces excessive activation charge density.

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C. Analysis of Crack Growth

Continued operation of a reactor in which defects due to IGSCC are known to exist must be based on an adequate demonstration of the safety of the reactor in the presence of a flaw. This analysis requires an assessment of the margin to failure of the flawed piping and an analysis of the possible nature of the failure, i.e., a leak or rapid catastrophic rupture. Linear elastic fracture mechanics (LEFM) has been extremely successful in determining the effects of a flaw on the performance of a structure for materials of relatively low toughness. However, the carbon and stainless steels normally used for reactor piping systems are generally very tough at the temperatures of interest, and the development of fracture mechanics techniques adequate to describe the failure of tough ductile materials is not complete.

The NRC and EPRI, the Central Electricity Generating Board (CEGB) in the UK, and, to a lesser extent, the National Science Foundation and the Department of Energy have for several years sponsored work leading to an elastic-plastic fracture mechanics methodology which includes criteria for the onset of crack extension, for stable crack growth, and for final instability.* The sponsored work includes theoretical and analytical development and study of elastic-plastic fracture concepts, development of testing techniques, experimental verification of concepts, and development of design engineering procedures based on elastic-plastic fracture mechanics. It should be noted that a complete theory of the effect of IGSCC on structural performance requires an understanding of IGSCC crack initiation and crack growth and criteria for initial crack extension, stable crack growth, and final instability under purely mechanical loads. Even in the LEFM case a completely satisfactory theory is not available, and the major efforts in the past few

*A listing of recent and ongoing research programs is given in the Appendix.

years have been directed toward the development of elastic-plastic failure criteria under mechanical loading rather than environmentally assisted crack initiation and growth. A comprehensive review of elastic-plastic fracture mechanics methodology is beyond the scope of this report, but a brief review of current work is helpful in developing an overview of the cracking problem.

1. IGSCC Initiation

From a phenomenological viewpoint, the methodology employed in Section III of the ASME Code for design against fatigue crack initiation seems reasonably satisfactory when applied to well-characterized systems. The fatigue failures encountered in LWR piping systems do not appear to have arisen because of inadequacies in the design methodology, but rather because additional stresses were present that were not accounted for in the original design. No similar design procedure is available for SCC.

A "stress rule" has been proposed by GE[1] for assessing the vulnerability of BWR welds to IGSCC. This rule is based on the premise that IGSCC will not occur if the sustained stress at the weld does not exceed the 0.2% offset yield strength at the operating temperature. Expressed in terms of stress quantities calculated in an ASME Code type of analysis, the stress rule requires

$$\frac{P_m + P_b}{S_y} + \frac{Q + F + R}{S_y + 0.002E} \leq 1 ,$$

where P_m is the primary membrane stress, P_b is the primary bending stress, Q is the secondary stress, F is the peak stress, R is the residual stress, S_y is the yield stress at operating temperature, and E is Young's modulus. Since the

secondary, peak and residual stresses in a Code analysis are calculated on an elastic basis, the stress rule treats these stresses in terms of the strains associated with them and requires that the total strain corresponding to the secondary loads not exceed the yield value.

It is not clear that the fundamental premise of the rule is correct, nor is it clear that the way loads are combined is meaningful in the highly thermoplastically deformed HAZ; however, application of the rule to field cases of IGSCC suggests that to date it has been somewhat successful in screening those welds that have not yet cracked. It is less useful for ranking the vulnerability of welds having an index greater than 1.0 (50% of BWR welds). Less than 1% of the welds in service have actually cracked, and there is no obvious correlation between the probability of cracking and the magnitude of the stress-rule index.

The development of an improved stress-rule index is currently being studied under EPRI sponsorship.[2] This work attempts to extend an analysis of the CERT test, using the film-rupture model of IGSCC applied to sensitized stainless steel by Ford and Silverman,[3] to arbitrary loading histories.[4] This leads to a cumulative damage criterion for failure which depends explicitly on loading history and time. The initial results[4] are interesting and suggestive, but more data are needed to define the parameters of the model and to verify the basic hypotheses.

2. IGSCC Growth

Once a crack has initiated, understanding crack growth is crucial both in determining an appropriate course of action and in evaluating the consequences of failure, i.e., leak-before-break or catastrophic rupture.

Relatively few measurements of crack growth rates for sensitized Type 304 stainless steel are available, although work is in progress in at least three EPRI programs[5-7] to obtain crack growth rates as a function of stress intensity level, sensitization, and oxygen level. Early results[8] gave very high crack growth rates [$\dot{a}$ = 200 mm/yr for a crack-tip stress intensity K = 16 MPa(m)$^{1/2}$] in a simulated BWR-type environment (280°C, 8 ppm O_2); similar rates are obtained when average crack growth rates are estimated from CERT tests.[9,10] However, more recent work using fracture-mechanics-type specimens gives much lower values. Measurements[11] on bolt-loaded* 1T-CT specimens from two different heats of material which were furnace sensitized and exposed to high-purity water at 288°C with 8 ppm oxygen typically give $\dot{a} \sim 3$ mm/yr for $K \sim 20$ MPa(m)$^{1/2}$. For six specimens with measurements extending over six thousand hours, the growth rates ranged from 1 to 4 mm/yr for K values in the range 18-24 MPa(m)$^{1/2}$. However, these tests were interrupted to make optical measurements of the crack length, and consequently, these rates may underestimate the true rates because of the time lost in reinitiating the crack after the interruption. (This may not be a serious problem; growth rates measured over two 1,000-h periods are in good agreement with rates measured over a single 2,000-h period.) Preliminary data from other investigators[12] also indicate that the crack growth rates obtained from fracture-mechanics tests are much less ($\dot{a}$ < 25 mm/yr) than the estimates obtained from CERT tests. Fracture-mechanics-type crack-growth tests are also planned for specimens from actual weldments. Tests are planned to determine crack growth rates in the HAZ[7] and in the weld metal itself.[12]

It is not clear that K is a suitable variable to characterize IGSCC growth in Type 304 stainless steel, since relatively large plastic zone sizes

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*The specimens are not conventional bolt-loaded specimens, but rather CT specimens which are placed in an external loading jig.

occur even at fairly modest load levels. There is some indication that even for CT specimens, crack growth rates can differ by an order of magnitude for the same K depending on whether the system is in load control or displacement control,[12] although part of the difference is undoubtedly due to heat-to-heat variations. Comparative tests on a single heat of material are under way at ANL.[5] In other work,[9] it has been found that small cyclic loads superposed on a nominally constant load significantly increase the crack propagation rate over that expected from simple addition of the rate associated with a constant load of the same magnitude and the rate associated with crack propagation under a purely cyclic load. Additional tests are needed on non-bending geometries like center-cracked panels and a variety of loading histories to further determine the suitability of K.

Analytical studies of IGSCC crack growth in weldments based on LEFM concepts are also being sponsored by EPRI.[13,14] The basic model is described in detail in Ref. 15. A residual stress distribution from an uncracked pipe is obtained from experiment. Since the problem is assumed to be linearly elastic, the solution for a cracked pipe under an applied load can be obtained by linear superposition of three simpler problems: a cracked pipe with a given applied loading, an uncracked pipe with the residual stresses acting on the plane of the crack, and a cracked pipe with applied tractions acting on the crack faces which are equal and opposite to the residual stresses.

In the model presented in Ref. 15, the redistribution of the initial residual stresses as the crack propagates is treated as a purely elastic problem. Subsequently, an analysis[16] was carried out which indicated that this is a reasonable approximation as long as the crack is less than half the wall thickness. The some-

what more general studies in Ref. 17 suggest a similar conclusion. Since the model predicts that most of the life of the cracked pipe is spent at short crack lengths, the redistribution may not greatly affect the lifetime calculations. However, although the results in Ref. 16 and 17 are suggestive, the behavior of the highly nonuniform residual stresses associated with weldments are much more complex than the situations modeled there, and the real significance of load redistribution is still an unanswered question.

Perhaps a more serious objection to the use of LEFM is the relevance of a crack growth rate law based on a characterization of the stresses near the crack tip in terms of the elastic crack-tip stress intensity factor K to crack growth in a material subjected to extensive thermoplastic deformation. Linear superposition of the residual stresses and the operating stresses for most piping systems will give stresses above the nominal yield. It can be argued that the loading is actually occurring along the linear reloading portion of a work-hardened stress-strain curve. In this case superposition and characterization of the crack-tip stresses by K could be justified, but the relationship between $\dot{a}$ and K in the inhomogeneous thermoplastically deformed HAZ may well be different from that measured on laboratory specimens.

In view of the many uncertainties involved in nearly every phase of the analysis, it is unlikely that accurate lifetime predictions can be made for real pipes. However, a fracture-mechanics model may be able to give reasonable bounds on lifetimes. Comparisons of the analytical predictions can be made with tests that have been carried out on precracked carbon steel pipes,[18] and with ongoing[19] and planned[6] tests on precracked Type 304 stainless steel pipes.

Although the characterization of fatigue crack growth in terms of LEFM concepts is well developed [see, e.g., Ref. 20], and a data base exists for fatigue crack growth rates in LWR environments,[21-25] as noted in Ref. 26, there are serious shortcomings when this methodology is applied to reactor piping systems. Fundamentally, LEFM assumes that the plastic zone associated with a crack tip must be small with respect to the crack size and the remaining ligament. Since pipe stresses near yield can be expected in some transients, and in many cases the total applied stresses will be greater than 50% of yield, the *a priori* applicability of LEFM is not obvious. Despite these shortcomings, the LEFM concepts represent the state of the art and are a vital ingredient in current failure assessments [see, e.g., Ref. 26]. There is also evidence to suggest that the LEFM crack-growth relations do provide useful correlations at ΔK levels well beyond those for which LEFM concepts would be considered valid.[27] Currently, cyclic crack growth rate studies in structural materials in prototypical environments are being sponsored by both the NRC and EPRI, but the general framework for understanding and interpreting the measurements is still LEFM.

Experimental verification of LEFM predictions in materials and structural configurations typical of those encountered in reactor applications is limited. In work sponsored by the CEGB,[28] an experimental study of crack shapes has been attempted. Axial starter cracks were machined in 6-in.-dia piping. The flaws were long compared to the thickness ($2c > 15t$) and deep ($a \lesssim 0.5t$). Fatigue tests under internal pressure were then carried out to determine the final shape of the crack when it penetrated the wall. In these tests the wall penetration was limited to, typically, 20% of the initial defect length. No axial growth was observed. Again it should be noted that until a

satisfactory model for cyclic crack growth in ductile materials at loads typical of reactor systems is established, it is not clear how small-scale experiments can be interpreted in terms of actual piping systems.

3. Failure Criteria and Leak-Before-Break

Recent experimental, computational, and analytical studies[29-33] have contributed substantially to the characterization of elastic-plastic crack growth. Probably the most systematic assessment of candidate fracture criteria was carried out in the EPRI-sponsored programs at the General Electric Corporate Research and Development Center[34] and at Battelle-Columbus.[35] Experiments were performed on A533B steel and "toughness-scaled" 2219-T87 aluminum plate (the toughness scaling was accomplished by preserving the ratio between the plane-strain plastic zone size and the plate thickness of a nuclear pressure vessel). Tests were done on compact specimens and center-cracked panels with varying sizes of remaining ligaments. The records of experimentally measured load (or displacement) versus crack growth were then used as input to finite-element analyses of the experiments. The dependence of potential fracture criteria on crack extension was then observed when crack growth in the finite-element model was forced to follow the measured relations between crack extension and load (displacement). The process was then inverted; a fracture criterion was chosen and a finite-element analysis was used to predict the crack growth behavior of a specimen under load, and experiments were then performed to verify the predicted results. These studies indicated that in many cases the crack growth behavior of ductile materials can be characterized in terms of J- or COD-resistance curves (J-R or δ-R curves).

Of the two approaches, J is the more convenient from a computational and analytical viewpoint because of its history independence and its relative insensitivity of finite-element size and type. However, since it is based on deformation plasticity, it is inadequate when large amounts of stable crack growth accompanied by large-scale plasticity occur. In such cases, J is no longer specimen independent.[30] A fracture criterion based on COD[29] (measured at a fixed distance behind the current crack tip) or equivalently a crack tip opening angle (CTOA) that reflects the actual slope of the crack faces[30] appears to be more satisfactory in this case, although computationally less convenient and difficult to measure experimentally.

Theoretical and experimental support for the J-k curve approach has also been provided by Paris and his coworkers in a program largely supported by the NRC.[31] In addition, this work has developed a simplification of the basic J-R curve approach, in which the toughness properties of ductile metals are characterized by only two parameters. The material toughness associated with initiation is assumed to be characterized by J_{IC} (or, equivalently, δ_{IC}, the crack opening displacement), and the material toughness associated with crack growth is assumed to be characterized by the dimensionless parameter $T = (E/\sigma_Y^2)(dJ/da)$ [or, equivalently, $T_\delta = (E/\sigma_Y^2)(d\delta/da)$] where E is Young's modulus and σ_Y is the yield stress, and T is referred to as the tearing modulus. Since in many cases the tearing modulus can be readily estimated, this approach can be useful for simple assessments of crack-growth instability. It should be noted, however, that the validity of this approach is subject to the same small crack-growth limitations as the J-R curve approach. The linear representation of the J-R curve also frequently overestimates the fracture resistance of the material.

Current research funded mainly by the NRC and EPRI is focused on three areas: (a) The development of efficient test procedures for the determination of J-resistance curves and the measurement of J-R curves for the alloys of interest in nuclear reactor primary systems, (b) experimental verification of current elastic-plastic fracture methodology on structural components, and (c) development of analysis techniques to apply elastic-plastic fracture methodology to actual structural systems.

There is still a need to develop standardized test procedures for obtaining J-R curves. The NRC is sponsoring research to explore various promising procedures for steels (A106, A516, A351-CF-8A) and weldments. Irradiated material will also be examined. The EPRI-sponsored programs put less emphasis on the development of test techniques, but the determination of J-R curves for carbon and Type 304 stainless steel piping is included.

Quasi-static loading tests on through-wall circumferentially cracked Type 304 stainless steel pipes[36] show that crack initiation can be predicted by a net section flow stress criterion. Because initiation is nearly coincident with maximum load, this criterion can also be used to predict fracture under load-controlled conditions. Tests on center-cracked panels indicate that a net section flow stress criterion can even be used for more complex types of loading (e.g., cyclic or seismic loads).[37] Pipe experiments with initial partially through-wall cracks give failure loads that are somewhat larger than the net section flow stress criterion would predict. However convenient for some design purposes, a net section collapse approach is inapplicable for some problems such as instability under nearly displacement-controlled conditions.

Programs on the experimental verification of current elastic-plastic fracture-mechanics methodology are being sponsored by both the NRC and EPRI. Tests will be carried out on both laboratory specimens (with variations in specimen type and geometry and the associated overall system stiffness) and piping components. Both static and dynamic loading will be considered. The pressure-vessel tests being carried out under the HSST programs also provide an opportunity for the verification of this methodology.

A principal impediment to the implementation of elastic-plastic fracture mechanics to structural situations is the lack of reliable calculations of J (or δ) for typically encountered flaw configurations such as the part-through surface crack. The simplified analyses based on the tearing-modulus concept developed chiefly by Paris and his co-workers (see, e.g., Ref. 37) have shown that the methodology of elastic-plastic fracture mechanics can yield insights into the behavior of actual structural components, although it can be difficult to assess *a priori* the degree of conservatism inherent in highly simplified models.

The development and assessment of simplified analysis procedures are being addressed by both EPRI- and NRC-sponsored programs. The tearing-instability analysis of piping[38] will be examined to ensure the reasonableness of its assumptions and to determine its range of validity. EPRI-sponsored work at GE[39] is also aimed at developing a "handbook" approach to elastic-plastic fracture based on detailed finite-element analyses of commonly occurring flaw configurations. This program will also develop techniques to determine an equivalent compliance for piping systems.

It should also be noted that current elastic-plastic fracture mechanics techniques are generally applicable only to two-dimensional problems. To analyze real problems, it is necessary to develop an idealized two-dimensional problem that will lead to conservative predictions of failure. Such a procedure may be overly conservative in some cases and, even more importantly, difficult to do in other cases.

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D. Nonenvironmental Corrective Actions

1. Introduction

The fundamental premise of the current efforts to prevent IGSCC in BWR piping is that IGSCC involves a complex interaction between material susceptibility (sensitization), the stresses acting on the material, and the environment, and that by suitably altering or varying these parameters immunity from IGSCC can be produced. The effect of environment on IGSCC will be discussed in more detail in a later section. However, it should be noted that although there is some evidence that the higher oxygen levels (up to 8 ppm O_2) associated with start-ups and shutdowns serve to initiate IGSCC,[1] later work[2] has shown that susceptible material can crack in the normal operating environment (0.2 ppm O_2); in fact, it appears that the threshold oxygen level for cracking of sensitized Type 304 stainless steel is ~40 ppb.[3,4] Hence, both the start-up and operating BWR environments are sufficiently aggressive to promote IGSCC in susceptible materials.

Nonenvironmental corrective actions seek to mitigate either the material susceptibility or the state of stress on the inside surface of the weldment. They include techniques for improving the margin against IGSCC of a susceptible material like Type 304 stainless steel and the identification of alternate materials that are inherently more resistant to IGSCC. In this country most of the work on these corrective actions has been sponsored by EPRI or by the BWR Owners Group organized by EPRI; a listing of the ongoing research programs is given in the Appendix.

The most extensively studied techniques for improving the IGSCC of Type 304 stainless steel include solution heat treatment (SHT) of pipe welds, corrosion-resistant cladding (CRC) of the weld HAZ, heat-sink welding (HSW), and induction-heating stress improvement (IHSI).

Operating experience and laboratory testing indicate that virtual immunity of Type 304 stainless steel to IGSCC can be produced by eliminating weld-sensitized regions. This can be done by SHT to redissolve the chromium carbides which have precipitated and eliminate the chromium-depleted regions. In addition, SHT will relieve cold work and weld-induced residual stresses. After welding, GE[5] now recommends an SHT in which the entire pipe segment is solution annealed at 1038° to 1148°C (1900° to 2100°F) for 15 min per inch of thickness, but not less than 15 min nor more than 1 h regardless of thickness, and then quenched in circulating water to a temperature below 204°C (400°F). The use of SHT is limited, however, to those weld joints made in shops where heat-treatment facilities are available and to piping assemblies physically small enough to be accommodated by a furnace and quench tank.

Corrosion-resistant cladding is really the application of an alternate material to protect the susceptible HAZ of the weldment. Type 308L weld metal is applied as a weld deposit to the inner surface of the pipe at the pipe weld ends before the final field weld is made. The overlay covers the region which will become sensitized when the final butt weld joint is made. Both field experience and laboratory studies[6,7] suggest that Type 308 and 308L weld metals with sufficient ferrite levels are very resistant to IGSCC in BWR environments [although a recent instance of IGSCC propagation through Type 308 weld metal with a fairly high (>5%) ferrite content has been reported].[8] For Type 308L weld metal, the minimum ferrite level currently recommended for CRC is 8%.[3] In practice, two different CRC procedures are used depending on whether or not it is possible to solution heat treat the

piping after the cladding is applied. The two different procedures have been referred to as "shop" and "field" CRC. In the shop CRC procedure, a region of cladding is first applied away from the weld end, which produces a sensitized region on the inner surface of the pipe adjacent to the weld deposit. The pipe is solution heat treated to eliminate this sensitized region, and then the remaining cladding is applied out to the weld end. In the field CRC procedure, the cladding is deposited without SHT, which leaves a sensitized zone on the inner surface of the pipe, but it is far from the region of high residual stresses associated with the butt weldment. Both SHT and CRC have been recognized by the NRC, in NUREG-0313, as acceptable approaches for producing IGSCC-resistant piping systems. However, CRC does increase the difficulty of carrying out reliable ISI.

Substantial experimental[9,10] and analytical evidence exists to show that very high axial tensile residual stresses are present on the inner surface of piping in the HAZ of a conventional weldment. These residual stresses and the associated sensitization increase the susceptibility of the weldment to IGSCC. In the HSW approach, the inside surface of the pipe is water cooled during all weld passes subsequent to the root pass or the first two layers. Experimental data[3,11] indicate that HSW does substantially lower the degree of sensitization associated with weldments in a given heat of material and lowers the tensile residual stresses on the inner surface (in many cases, the inner surface is left under compressive stress). However, although sensitization levels are reduced by this process, susceptible heats of Type 304 material would still have sensitization levels higher than the current estimate[12] of the "threshold level" for a BWR environment. It

appears that the effects of HSW on residual stress are more beneficial for small-diameter pipes than for large-diameter pipes because of the greater thermal inertia of the latter.

The SHT, CRC, and HSW remedies are applicable to Type 304 stainless steel in field repair or replacement and in the construction of new plants. To mitigate cracking in existing plants, modification of the residual stress state near the weldment has been proposed. To accomplish this, the pipe weld joint is induction heated from the outside while the inner surface of the weldment is water cooled. Typically, the outer surface of the pipe reaches ~500°C while the inner surface remains at <200°C. This IHSI process produces a throughwall residual stress distribution that tends to increase resistance to IGSCC. Although the plastic strains induced during the induction-heating cycle may affect the susceptibility of the weld to low-temperature sensitization, in the short term the relatively low temperatures involved have no effect on the degree of sensitization, and the IHSI process thus appears to affect IGSCC behavior only through the changes it produces in the residual stress state near the weldment.

The IHSI process has been extensively developed by the Japanese.[13-16] In addition to developing the technique and studying the proper selection of process parameters, they have measured residual stresses on the inner surface of weldments after IHSI. The weldments examined[14] included different types (pipe to pipe, pipe to reducer, pipe to elbow, etc.) representative of the joints encountered in an actual reactor piping system, including large-diameter (20-in.) weldments. In most cases, the IHSI treatment did produce compressive residual stresses on the inner surface, although, at least on the basis of the work reported in Ref. 14, the procedures and process parameters

have not been optimized for all configurations to ensure consistently satisfactory results. As in the case of HSW, it also appears that IHSI is more effective for small-diameter than for large-diameter pipes, again because of the greater thermal inertia of the latter.

SHT, CRC, HSW, and IHSI are all intended to improve the margin against IGSCC in Type 304 stainless steel. Another approach applicable to repairs or new construction is to use an inherently less susceptible material. Candidate alternate materials include carbon steel*, Types 304 and 316 "Nuclear Grade" stainless steels, Types 304L and 316L (low carbon) stainless steels, Type CF-3 cast stainless steel, Type 347 stabilized stainless steel, and XM-19 stainless steel. The "Nuclear Grade" materials are ASME Type 316 or 304 materials with carbon content of 0.02% maximum and controlled nitrogen content in the range 0.06 to 0.10% maximum. These meet the ASME requirements for maximum carbon and nitrogen content of 0.08 and 0.10%, respectively. The low carbon levels reduce susceptibility to sensitization and hence increase resistance to IGSCC. By virtue of nitrogen contents in the high end of the allowable range, these materials meet the strength specifications for Types 304 and 316 stainless steels. This allows one-to-one replacement of material in existing plants or those under construction without redesign or additional stress analysis.

2. Residual Stresses in Pipe Weldments

Both the IHSI and HSW remedies depend primarily on the modification of the residual stress state on the inner surface of the weldment. Much of the

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*Carbon steel is not an alternate material in the sense that it can replace Type 304 stainless steel everywhere in the reactor piping system. However, since in later-model BWRs (BWR-5 and -6) large portions of the piping are carbon steel, it is important to quantify its resistance to environmentally assisted attack.

available US data on welding residual stresses produced by standard nuclear industry practices are summarized in Refs. 9 and 10. Butt welds were made on 4-, 10-, and 26-in. Schedule 80 Type 304 stainless steel piping. In these two programs, four 4-in., two 10-in., and two 26-in. weldments were examined. The Japanese have also made extensive residual-stress measurements; typical results are given in Ref. 11. The available data indicate that residual stresses vary in both the axial and azimuthal directions, although the distributions in the 26-in. weldments appear to be reasonably axisymmetric. Both axial and circumferential residual stresses are present, but since circumferential cracks can grow and remain in the sensitized region, while axial cracks grow into the weld metal or unsensitized regions which tend to resist IGSCC growth, the axial residual stresses appear to be the more important for IGSCC.

Finite-element calculations[17] of the residual stresses suggest that the peak tensile residual stresses decrease with increasing pipe size. There is a wide scatter between nominally identical weldments in the experimentally measured values of residual stress, and the data base is too limited to draw any firm conclusions, but based on both the Japanese and the U.S. results there does seem to be some tendency for the tensile residual stress to decrease in large (>20 in.) pipes. The peak tensile stresses from the US tests in the heavily sensitized region 2-3 mm from the weld fusion line are given in Table III.2.

It should be noted that the stresses in Table III.2 are primarily associated with the thermomechanical cycle associated with welding. To obtain good alignment of the inner surfaces in a pipe-to-pipe joint, it is usually

Table III.2. Peak Axial Tensile Stresses

Pipe Size	Ref. 9	Ref. 10	Ref. 18
4-in.	260, 320, 350 MPa[a] (38, 46, 51 ksi)	275 MPa (40 ksi)	-
10-in.	415 MPa (60 ksi)	205 MPa (30 ksi)	-
16-in.	-	-	345 MPa (50 ksi)
26-in.			
High Heat Input*	195 MPa (28 ksi)	205 MPa (30 ksi)	-
Low Heat Input	-	105 MPa (15 ksi)	-

[a]Three tests performed.

necessary to machine a counterbore on the ends to eliminate the effects of any out-of-roundness or thickness variations of the pipe. The counterboring is usually done by relatively rough machining which can lead to high residual stresses on the counterbore surface. As reported in Ref. 9, these stresses seem to be annealed out during the welding process, at least in the weld-sensitized region, and hence pre-weld surface treatment does not appear to play a significant role in IGSCC. However, studies[10] indicate that post-weld grinding can lead to very high [up to 550 MPa (80 ksi)] surface residual stresses. Although these surface residual stresses extend only a few mils into the wall, they appear to play a significant role in IGSCC initiation.[3] Japanese measurements of welding residual stresses generally are higher than

*The measurements reported in Refs. 9 and 10 were made at different azimuthal positions on the same weldment.

those obtained in the US; their welding practice appears to use substantially higher heat inputs after the first one or two passes than does US practice.[10,11]

It seems clear that HSW and IHSI can in many cases either reduce the tensile residual stresses or even leave the inside surface completely in compression. Japanese studies on HSW[11] and IHSI[13-16] which support this conclusion have been cited previously. In this country, throughwall residual stresses have been measured on a 4-in. weldment[18] which was given an IHSI treatment after a conventional weld procedure and post-weld grinding. This produced strongly compressive stresses on the inner surface.

One variant of HSW that is of particular interest in the EPRI program is the so-called last-pass HSW procedure in which inside cooling is applied only during the final, very high heat-input welding pass. Finite-element calculations[17] suggest that this approach produces a strongly compressive stress state on the inner surface of the pipe (although it is probably not effective in reducing weld sensitization). Experimental results have been obtained on a 12-in. weldment[19] where extremely compressive residual stresses [$\sim$ -240 MPa (-35 ksi)] were obtained on the inner surface. A similar procedure on a 28-in. weldment[19] left a small portion on the inner surface with modest tensile stresses [$\sim$35 MPa (5 ksi)]. It is possible that a variant of last-pass HSW could be utilized as an alternative to IHSI. A portion of an existing weld would be removed and an additional layer of weld metal deposited during a high-heat-input pass with water cooling on the inner surface. Since the thermal gradients produced would be similar in character to those produced by IHSI, a decrease in the tensile residual stresses on the inner surface would be expected.

The exact role of residual stresses in IGSCC in an operating reactor is not clear. IGSCC produced solely by welding residual stresses has not been reported in simulated BWR environments, although such residual stresses will readily produce cracking in more aggressive environments (e.g., $MgCl_2$). Other tests indicate that in BWR-type environments tensile (primary loading) samples fail much more readily than bent-beam (secondary loading) samples with nominally equivalent loads.[20] It also can be argued that if the residual stresses provide the crack driving force, then a small amount of crack growth will relax the residual stresses, thus relieving the crack driving force. It is certainly true that sufficient crack growth will cause redistribution and relaxation of the residual stresses, but the current best estimate[21] suggests that significant redistribution will not occur until the crack grows through a significant fraction of the thickness. Perhaps more critical to an understanding of the detrimental effect of the tensile residual stresses due to normal welding practice and the possible beneficial effects of the stress states produced by HSW and IHSI is the influence of reactor operating loads on the residual stresses near the weld. Some Japanese data are available[13] on the effect of load cycling on the state of residual stress near weldments. These results suggest that for loads less than 0.8 of the nominal yield strength, little relaxation occurs; however, for loads larger than yield, significant relaxation occurs, and for loads equal to 1.2 times yield almost complete relaxation occurs.

Operating stresses are limited by the code under which the piping system is designed. The ASME Boiler Code, Section III, "Nuclear Power

Plant Components," is the current code used for the design of piping systems in nuclear power plants, although many operating reactors have been built using ANSI B31.1 piping standards. For the purpose of defining design stress limits, stresses are classified into different categories depending upon their nature and origin. The classification is generally based upon the concepts of elastic-perfectly plastic limit analysis (see, e.g., Ref. 22). Primary stresses are produced by the imposition of external loads, e.g., pressure, dead weight, and seismic forces and moments. The deformations produced by these loads are not self-limited. Secondary stresses arise as a result of the displacement constraints placed on the structure. The major characteristic of these stresses is that the deformations associated with them are self-limiting, i.e., even if the structure yields, it displaces only by the amount necessary to satisfy the displacement constraint. Examples of this type of stress are thermal and discontinuity stresses. The codes do not account for residual stresses associated with welding, fit-up, etc. The primary membrane stress is limited to two thirds of the yield strength at temperature. However, the code-specified limit on the primary plus secondary stresses (calculated on an elastic basis) is approximately two times the yield strength. Since the code design rules only place an upper limit on allowable stresses, it is likely that most welds will not have operating stresses that exceed the yield strength. Although the code limits on combined primary plus secondary loads are within the shakedown limit of an elastic-perfectly plastic material, the effect of such loadings on the residual stresses introduced by prior plastic deformation is not clear.

Experimental studies on residual stresses in weldments from operating plants are limited. Two 4-in. pipe-to-elbow weldments and a 10-in. pipe weldment that were removed from service were examined in Ref. 9. The 10-in. weldment had very high residual stresses, and the 4-in. weldments had relatively low tensile stresses. However, the actual histories of the weldments are not known, and it is possible (although metallurgical examination suggests that it is unlikely) that the 4-in. weldments were stress-relieved (annealed). Analytical studies of the stress redistribution of IHSI-treated pipes under experimental testing conditions and typical in-service operating conditions are planned by EPRI.[23]

3. Qualification of Proposed Remedies and Alternate Materials

In the initial development and discussion of proposed remedies and alternate materials, the focus has been on reducing the severity of the stress levels or the sensitization. However, since a quantitative understanding of the role of stress or sensitization in IGSCC does not exist, the actual benefits of each remedy must be assessed through actual SCC susceptibility tests. The chief procedures that have been used to evaluate the effectiveness of the remedies are the CERT test, full-scale laboratory pipe tests, and small-specimen uniaxial tests.

The CERT test (see, e.g., Ref. 24) involves the application of a relatively slow nominal strain rate ($\dot{\varepsilon} = 7.3 \times 10^{-7}\ s^{-1}$ is a typical value for many of the tests used to screen remedies and alternate alloys in the EPRI-GE program) to a test specimen exposed to an environment. While the expression "constant strain rate" is common usage, the tests are actually conducted with a constant crosshead speed, and the actual strain rate will vary,

especially after cracks begin to propagate. With a proper choice of strain rate, susceptible materials in aggressive environments undergo SCC which is metallographically indistinguishable from that produced in constant-load experiments. The test is concluded when the specimen fractures and the crack susceptibility is then related to the mode of fracture, either ductile fracture or SCC. The susceptibility is measured by visual evidence (indications of IGSCC), time to failure, and the reduction in area. The CERT test is a relatively severe type of test in the sense that it promotes stress-corrosion failure in relatively short times compared, for example, with constant-load tests. It is essentially a test of material susceptibility in an environment under loads that can reach the ultimate tensile strength of the material (if no cracking occurs). It should be noted that the degree of IGSCC observed in a CERT test can be strongly dependent on strain rate,[25] and tests at a single strain rate may give an incorrect picture of IGSCC susceptibility.

The CERT test has been used primarily as a screening test in the programs to evaluate remedies for pipe cracking in BWRs. In order to relate laboratory results to operating experience, a uniaxial "pipe test" which more closely simulates operating conditions was developed. The pipe testing procedure[3] consists of stressing welded pipes in a cyclic tension mode. High-purity water is circulated through the pipe during the test. The load in most tests is increased linearly over a 5-min period until the stress reaches a value of

approximately 136% of the 288°C yield strength* of the particular heat being tested, held at that level for 75 min, reduced linearly to a value of approximately 10 MPa (1.5 ksi) over a time period of 0.5 min, and held at that level for 5 min. This history is then repeated for subsequent cycles. For most tests the environment is high-purity water at 288°C (550°F) with 8 ppm dissolved oxygen, at a flow rate of 1.3-1.9 x 10^{-4} m^3/s (2-3 gpm) and an internal pressure of 8 MPa (1120 psi). (The 1.36 σ_y applied stress is applied by an external hydraulic system. There is an additional 20 MPa (2.9 ksi) stress due to the internal pressurization; thus, the total applied stress is approximately 1.5 σ_y.) The electrical conductivity of the water is maintained below 1 μmho/cm. The test weldments are fabricated by butt welding eleven 100-mm-long segments of 4-in. Schedule 80 pipe together to form a 1.1-m-long test section. Each end of the test section is welded to a transition piece fabricated from 4-in. Schedule 160, Type 316 stainless steel pipe. The transition pieces are welded to end-loading adapters which seal the specimen and are in turn connected to the external hydraulic loading system.

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*The particular choice of 136% of the yield strength appears to be a peculiarity of the history of the development of the pipe test. The first tests were carried out in bending, and the maximum stress (calculated on an elastic basis) for a 4-in. Schedule 80 pipe under a plastic limit bending moment is 1.36 σ_y. This was then carried over to the uniaxial tests. However, it should also be noted that 1.36 σ_y is just slightly greater than 2 S_m where S_m is the code design stress level. The maximum stress in the pipe test then exceeds the limit for primary membrane stresses (S_m) and the limit for primary membrane plus bending stress (1.5 S_m), but it is less than the limit on primary plus secondary stresses (3 S_m). Choosing the applied stress as a fixed percentage of the yield strength was an attempt to normalize results between heats of material with different yield strengths. Statistical analysis[26] of the data indicates, however, that the time to failure is a function of the absolute value of the applied stress. New pipes in the alternate alloy program conducted by EPRI/GE will be tested at 33.8 ksi, which is 2.0 S_m for Type 304 stainless steel where S_m = 16.9 ksi is the design stress level at 288°C specified by the ASME Section III rules for Class I components.

There has been a limited parametric test program[27] to investigate the effect of changing stress levels, dissolved oxygen content, and cyclic frequency. A few of the pipe tests for the HSW remedy were run with applied loads of 1.10 σ_y, and almost all the pipe tests for the IHSI will be run with lower applied loads.[18] As noted previously, Japanese studies[13] indicate that the stress levels normally used in the pipe tests at GE will relieve the residual stresses introduced by the welding process or by HSW or IHSI.

Evaluation of the susceptibility to IGSCC in the pipe tests is based on a comparison of times to failure or cycles to failure under the standardized loading history between reference specimens of welded Type 304 stainless steel and a candidate remedy. To qualify remedies intended to increase the margin against IGSCC in Type 304 stainless steel, i.e., SHT, CRC, HSW, and IHSI, the performance of the reference and candidate remedy was compared for the same heats. To qualify alternate materials of different compositions, the performance of many heats (typically 5-7) of the alternate material was compared with that of many heats of Type 304.

The goal of the statistical pipe tests conducted by EPRI and GE has been to establish a statistically significant (90% confidence level) factor of improvement for the remedies over the reference Type 304 weldments. Since the direct determination of a mean time to failure for a successful remedy would involve prohibitively long testing times, the tests were carried out only for a specific test time or until first failure of the remedy. The factor of improvement was then estimated from a statistical analysis of the times to first failure under the assumption that the variance of the logarithm of the time to failure of the remedy is not greater than that obtained for the reference material. A detailed description of the statistical analysis used to evaluate the test results is given in Ref. 29.

The factor of improvement chosen for the increase in mean failure times in testing for the remedy over the reference weldments was based on the mean time to failure for Type 304 weldments in service. The median service life for weldments which have cracked in operating reactors as of September 1979 was 56 months.[25] Certain lines and systems have shown a more pronounced tendency to crack and have accounted for the majority of such incidents. The systems with the highest incidence of cracking up to 1979, the recirculation bypass and core spray systems, have a median time to cracking of 33 months. It should also be noted that the mean time to cracking for large-diameter (>12-in.) piping appears to be much longer than that for smaller-diameter lines.[28] To provide assurance of a 40-year service life, a factor of 20 improvement in the mean time to cracking was set as an initial goal for remedy development.

Since the pipe test is not prototypical in terms of loading history or oxygen levels, it is assumed that the effect of the differences in load levels, cyclic rate, chemistry variations, etc. on life is proportional for the reference weldments and the remedies so that the factor of improvement obtained under the test conditions accurately reflects the factor of improvement expected under service conditions.[3,29] Thus, e.g., if raising the oxygen level from 0.2 ppm to 8 ppm accelerates cracking more in Type 304 stainless steel than in an alternate material, the factor of improvement from the test will be nonconservative; if the change in oxygen level accelerates cracking more in the alternate material than in Type 304, the factor of improvement will be conservative. The basic assumption of the pipe-testing program is that for the remedies and alternate materials being considered, the probability that the factor of improvement obtained from the pipe tests is nonconservative is "insignificant",[3] although no detailed justification of

this assumption has been given. Some parametric studies to determine the effects of stress level, oxygen concentration, cyclic rate, and temperature on IGSCC in Type 304 weldments are being carried out,[30] but the only parametric studies planned for alternate materials appear to be off-normal water-chemistry (chloride and resin intrusion) tests.[26] The use of the factor of improvement obtained from the pipe tests to predict field experience also seems to be based on the implicit assumption that the reference weldments, which are Type 304 stainless steel with post-weld grinding, are typical of the weldments which have cracked in service. While post-weld grinding certainly has been frequently used, it is not clear how many of the 133 cracking incidents (through January 1979)[28] have been associated with post-weld grinding. It may be that the reference weldments used in the pipe tests represent a more susceptible condition than the typical field cracking case, and hence the factors of improvement obtained from the pipe cracking test would be nonconservative when applied to in-service piping. The question is not whether the pipe cracking tests can demonstrate an increased margin against IGSCC, but whether the factors of improvement obtained in the pipe tests give quantitative measurements of the improvement in service life.

The pipe testing program for the near-term remedies consisted of twelve different pipes. The factors of improvement obtained for each of the remedies are shown in Table III.3.

Table III.3. Factors of Improvement Obtained in Pipe Cracking Tests with Various Near-term Remedies

Specimen Number	Heat Number	Remedy	Factor of Improvement[a]
E13	M7616	SHT	$\geq$89.5
E14	M0063	SHT	$\geq$ 8.7
E15	454970	SHT	$\geq$ 3.0
E16	M7616	CRC Field	6.5*
E24	M0063	CRC Field	$\geq$ 7.5
E25	454970	CRC Field	1.9*
E32	M7616	CRC Shop	$\geq$88.1
E34	M0063	CRC Shop	$\geq$ 7.7
E35	454970	CRC Shop	$\geq$ 1.9
E19A	M7616	HSW 136% of σ	25.2 (3.8)*
E19B	M7616	HSW 136% of σ_y	15.1 (12)*
E20A	M7616	HSW 110% of σ_y	$\geq$ 7.7
E20B	M7616	HSW 110% of σ_y	(7.)*

[a] Only the specimens marked with an asterisk actually failed; the factors of improvement for the other specimens are based on the test time without failure. Values in parentheses are obtained by comparing the heat-sink welds with comparison weldments from the same specimen.

It should be noted that Heats M0063 (0.05% C) and 454970 (0.042% C) are resistant heats. The average time to first failure for the more than 10 heats studied in the EPRI/GE pipe-testing program is about 100 hours. The times to first failure for Heats M0063 and 454970 are 1067 and 3253 h, respectively. Thus, attempting to show a factor of 20 improvement would require very long-term testing and the relatively modest factors of improvement indicated by the SHT remedy for these pipes probably only reflect the relatively short test-program duration.

For the HSW specimens E19A and E19B, two factors of improvement are given. Specimen E19A contained 6 heat-sink welds and 6 standard welds, and the inside surface was polished before welding and received no surface treatment after welding. Similarly, E19B contained 6 heat-sink welds and 6 standard welds, and received the standard pre-weld machine "prep" and no surface treatment after welding. The factors of improvement quoted in Table III.3 are obtained by comparing the heat-sink welds with a post-weld ground specimen; the alternate factors of improvement in parentheses are obtained by comparing the heat-sink welds with companion weldments from the same specimen. Failure analysis[3] of specimen E20B indicates that the crack was completely transgranular, initiating at a machining groove at the end of the counterbore, and that it was probably due to fatigue rather than IGSCC. As noted previously, Ref. 13 indicates that residual stresses should relax in a relatively few cycles at the stress levels employed in these pipe tests. This suggests that the improvement observed in the pipe test is due primarily to the reduced sensitization associated with HSW. Pipe tests on IHSI (where any increased margin against IGSCC is due solely to the improved residual stress state) at 1.36 σ_y show little improvement over the reference weldments,[18,27] while

in tests at lower stress levels the IHSI treatment appears to increase the cracking margin.[31]

The factors of improvement obtained from pipe tests on the alternate alloys are shown in Table III.4. The "Nuclear Grade" Type 316 and 304 stainless steels and the Type 347 and CF3 stainless steels all appear to be much more resistant to IGSCC. Based on this improved behavior and other factors such as design/strength properties, ASME code status, weldability, inspectability, and availability, the Type 316 and 304 Nuclear Grade stainless steels were selected by EPRI/GE as the prime alternate alloys for BWR service,[26] and an expanded pipe-testing program to qualify these alloys for service is now being carried out.[32] The expanded program will include 3 heats of Type 316 Nuclear Grade material and 6 heats of Type 304 Nuclear Grade material plus 7 heats of Type 316L and Type 304L materials with a range of carbon, nitrogen, and molybdenum contents.[26] Roughly half of the specimens will receive an LTS treatment in addition to the standard welding plus post-weld grinding fabrication practice. (CERT tests on the Nuclear Grade materials with $\dot{\varepsilon} = 8 \times 10^{-7}\ s^{-1}$ at 288°C in high purity water with 8 ppm O_2 after an LTS treatment produced no IGSCC.[33]) The testing will be carried out until first failure or 5000 test cycles; if no failure occurs, this will correspond to a factor of improvement of ∿80 over the reference Type 304 material.[33]

Table III.4. Factors of Improvement Obtained in Pipe Cracking Tests on Alternate Materials

Alloy	Pipes Tested	Pipes Failed by IGSCC	Factor of IGSCC Improvement
Type 304	19	18	1
Type 316	3	3	∿1
Type 316 Nuclear Grade	11	0	≥51
Type 304 Nuclear Grade	8	0	≥48
Type 347	9	0	≥42
Type CF3	8	0	≥51

Prior to the introduction of the BWR-5 and -6 plants, carbon steel piping was confined primarily to the main steam and feedwater lines and to piping in the balance of plant. However, beginning in 1970, new BWR designs specified carbon steel in piping systems that were previously stainless steel. For the BWR-6 standard plant, except for the main recirculation water lines and parts of the control rod drive hydraulic system, virtually all piping systems are carbon steel. With the exception of erosion/corrosion failures and flow-induced vibration fatigue failures in the feedwater lines of some reactors, the operating experience with carbon steel piping and components has been excellent.[34] However, for many of the current applications of carbon steel piping, the actual number of years of service experience is fairly limited.

The current carbon steel testing program sponsored by EPRI at GE includes both full-scale pipe testing and small-specimen testing. The pipe tests are similar to those carried out for the austenitic stainless steels. The environment is 8 ppm O_2 water at 288°C; a similar cyclic loading history is used with maximum applied stresses ranging from 0.9 to 1.35 σ_y. Low-oxygen pipe tests with 0.2 ppm O_2 are also being conducted.

These tests have shown that environmentally enhanced cracking of carbon steels (SA 106-GrB, SA 333-Gr6) can occur in high-temperature oxygenated water. The cracks in the pipes are transgranular with flat fracture facets. The fracture morphology is typical of transgranular stress-corrosion cracking (TGSCC) in carbon steel.[34] For applied cyclic stresses of 135% of the 288°C yield strength, the time to failure for four pipes ranged from 190 to 875 h with an average time to failure of 525 h; the average figure for the Type 304 pipe weldments was 100 h for an applied cyclic stress of 1.36 σ_y. A

carbon steel specimen loaded to 1.25 σ_y lasted 1380 h; specimens loaded to 1.1 σ_y and 0.9 σ_y have lasted over 5800 h and 8700 h, respectively, without failure to date.[35] Pre-notched specimens (notch depths 0.009-0.08 in.) have failed in extremely short times. With applied stresses of 1.35 σ_y and 1.0 σ_y the failure times are 36 h and 45 h, respectively.[36] The pipe test data, together with small-specimen tests on low-cycle fatigue cracking, show that oxygenated-water environments accelerate crack initiation as well as crack growth rate.[36] In the pipe tests the cracks initiate primarily at geometrical discontinuities such as the counterbore or the weld fusion line. Environmentally induced cracking of carbon steel is not uniquely associated with the weld HAZ as is the case for sensitized stainless steel in high-temperature oxygenated water environments, and hence post-weld heat treatments do not produce beneficial effects. The weld area appears to act as a source of geometrical stress concentrations.

Instead of characterizing the applied cyclic stresses in terms of yield stress of the actual materials tested, one can characterize them in terms of the allowable design stress S_m permitted by the ASME code for Class I components. At 288°C, S_m = 18.1 ksi for carbon steel piping and 16.9 ksi for Type 304 stainless steel. Typical values of the applied stress for Type 304 pipe tests ranged from 1.7 to 2.2 S_m; for the initial carbon steel pipe tests, the applied stresses were approximately 2.6 S_m. Comparing lives based on stress levels expressed as a fraction of S_m, carbon steel pipe specimens appear to have times to failure 8 to 20 times greater than the Type 304 stainless steel pipe specimens under pipe-test conditions.

CERT tests (at $\dot{\varepsilon} = 3 \times 10^{-7}\ s^{-1}$) also indicate that environmentally induced degradation of carbon steel mechanical properties can occur in high-temperature oxygenated water. In contrast to sensitized Type 304 stainless steel, where the degradation is due solely to IGSCC, the degradation in carbon steel at oxygen levels typical of BWR operations is due to pitting at low temperatures ($\lesssim$150°C), to both pitting and transgranular cracking at intermediate temperatures ($\sim$150°C < T < $\sim$200°C), and to transgranular cracking at high temperatures.[36] A ductility minimum exists at approximately 232°C. At a given temperature, increasing oxygen from 0.2 ppm to 1.0 ppm accelerates pitting and TGSCC. For 8 ppm oxygen levels, crack penetration is more rapid as a result of pitting. The stress-corrosion behavior of carbon steel appears to be more sensitive to surface modifications and strain-rate variations than that of sensitized Type 304 stainless steel. At a fixed temperature, prefilming, surface punching, shot peening, or increasing the strain rate increases the degree of IGSCC.[35] Average crack-propagation rates can be estimated from CERT test results on shot-peened specimens where initiation is assumed to occur rapidly owing to mechanical rupture of the surface-hardened layer. For a given temperature and oxygen level, the average crack propagation rate for carbon steel (SA333-Gr6) is generally lower than that for sensitized Type 304 stainless steel.[35] However, for the nominal BWR operating conditions of 0.2 ppm O_2 and 288°C, the rates differ by less than a factor of two.[35] Tests on "statically" loaded (2.3×10^{-5} Hz) carbon steel compact tension specimens at 288°C in water with 8 ppm O_2 also show that crack propagation rates at high crack-tip stress intensities ($K = 40\ MPa \cdot m^{1/2}$) are similar to those in sensitized Type 304 stainless steel. However, the crack growth rate

in the carbon steel appears to decrease much more rapidly with decreasing stress intensity than in the case of sensitized stainless steel.[35,21] Ford[36] has argued that at high stress intensity levels, the rate-determining step in crack propagation at high temperatures in both carbon and stainless steel is liquid diffusion, and hence, that any process which hinders mass transport in the crack will decrease the crack propagation rate. Blockage by a voluminous corrosion product might be such a process in carbon steel. Thus under static loading conditions at lower crack-tip stress intensities with relatively small crack-opening displacements, the propagation rates in carbon steel should be considerably lower than those in stainless steel. Since cyclic loading produces a relatively high flux of fluid in the crack, corrosion fatigue would be expected to be a more common problem in operating plants than stress corrosion under static loading.

Cyclic crack growth rates[35,36] of carbon steel (SA 333-Gr6) in simulated BWR environments over a frequency range from 1.25 to 0.0125 cpm increase with increasing ΔK to values of $\Delta K \gtrsim 20$ ksi·in$^{1/2}$ (with $R = K_{min}/K_{max} = 0.5$). Above this value, the crack growth rate is independent of ΔK. The growth rate per cycle increases by a factor of four with each order of magnitude decrease in frequency. For $\Delta K \gtrsim 20$ ksi·in$^{1/2}$ the crack growth rates are, depending on frequency, $\sim$2-90 times those given by the ASME Section XI design curve for carbon steel in water. Since the ASME curve shows a monotonic increase in growth rate with ΔK with no plateau, it is conservative for higher ΔK. Other studies[34] on cyclic crack growth in SA333-Gr6 suggest that for a given ΔK the crack growth rate depends on the actual crack length. Thus for short cracks the crack growth rates obtained in Ref. 34 exceed the ASME curve,

while for the same ΔK the crack growth rate decreases at longer crack lengths. Better agreement between the data and the ASME curve is obtained[34] if ΔK is replaced by $\Delta K_{eff} = \Delta K\ (1-R)^{-0.5}$. The decreasing cyclic growth rates at increasing crack depths may be due to the "crack tip blockage" mechanism discussed earlier for the static crack-growth tests.[37]

Much more work is planned on understanding environmental effects on carbon steel piping. The ultimate goal of the EPRI program is a set of design stress rules to deal with environmental crack growth both in terms of initiation, as in the ASME Section III fatigue design, and in terms of a crack growth/fracture mechanics approach similar to the Section XI approach. At this time, however, it should be noted that based on the pipe testing carried out at GE, carbon steel is more susceptible to environmental degradation in BWR environments than other alternate materials such as the Type 304 and 316 Nuclear Grade austenitic stainless steels.

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E. Environmental Corrective Actions

Inhibitors are used extensively to mitigate corrosion and IGSCC problems in industrial processes and technological systems without significant modifications to the materials, stress state, or other conditions inherent in the particular application. In view of the effective utilization of inhibitors in engineering systems, this alternative is being evaluated in relation to IGSCC problems in the BWR along with the development of alternate materials and fabrication (e.g., welding) methods, and changes in the mode of operation to reduce the concentration of aggressive species in the high-purity water during start-up, shutdown, and lay-up of the system. Recent reviews[1,2] of the effect of chloride and oxygen on the SCC of austenitic stainless steels indicate that the oxygen concentration of the high-temperature water is the critical parameter and also a prerequisite for chloride SCC in this environment.

The criteria for an alternate water chemistry in BWR systems are quite restrictive in that most candidate inhibitors of IGSCC in austenitic stainless steels (1) are not compatible with the conditions and/or components in the reactor core or the steam turbine, (2) require extensive modification of the reactor coolant system, (3) impose prohibitive cost, or (4) present additional safety problems which must be resolved. Furthermore, a viable alternate water chemistry for normal operating conditions must be integrated with conditions and constraints imposed during start-up, shutdown, and lay-up of the system. In addition to controlling the oxygen concentration in the coolant by means of chemical additives during normal operation, it is also desirable to decresse the oxygen levels during the other modes of operation. For example, deaeration of the coolant during start-up has been demonstrated at the Vermont Yankee,[3]

Shimane,[4] Fukushima,[5] and other[6] nuclear power stations. Nitrogen blanketing and continuous deaeration of the condensate storage tank have been used in Sweden and Japan, respectively, to minimize oxygen contamination of the reactor coolant during the shutdown period. Implementation of deaeration and nitrogen blanketing of the coolant is, in part, based upon the observation that the frequency of pipe cracking in several BWRs is related to the number of reactor startups which followed cold-standby periods of $\gtrsim$ 24 h.[7] In the absence of contaminants from cation resin breakdown (viz., H_2SO_4), laboratory tests on tensile specimens under temperature- and stress-ramp conditions have demonstrated the benefit of startup deaeration relative to crack initiation and propagation in as-welded and as-welded plus sensitized Type 304 stainless steel.[8] Constant extension rate tests on sensitized Type 304 stainless steel also indicate that the average crack propagation rate increases as the temperature and oxygen concentration of BWR-quality water increase.[9] The latter work indicates that rapid SCC propagation can occur at low temperatures ($\sim$100°C) and oxygen concentrations of $\gtrsim$200 ppb; i.e., the conditions of temperature and oxygen concentration in BWR water during normal start-up conditions. There is also evidence that the threshold oxygen level for IGSCC of sensitized Type 304 stainless steel may be as low as 40 ppb at the normal operating temperature of $\sim$290°C. Although the time periods associated with start-up, shutdown, and normal lay-up are less than that during normal operation, the extent of degradation of the steel under the environmental conditions in each of the modes has not been established quantitatively. Consequently, improvement of the water chemistry during all modes of operation may be required to mitigate the occurrence of IGSCC of austenitic stainless steels in BWRs. Additional experimental results on the effect of O_2/H_2O_2 and other species on the corrosion potential and fracture behavior of sensitized stainless steel are required to establish the margin of performance of the material under simulated start-up, shutdown, and lay-up conditions.

Brown et al.[10] and Park et al.[11] have reviewed available information on the inhibition of SCC in alloys and metals of technological importance. Most of the results for austenitic stainless and ferritic steels were obtained in boiling magnesium chloride, strong acids, or caustic solutions and thus are of limited value in the selection of inhibitors for a BWR environment. Brown et al.[10] also investigated the inhibition characteristics of a number of additives in a standard 4N NaCl solution at 100°C by means of U-bend specimens, polarization measurements, pit initiation studies, and the rate of formation of films on a bare metal surface (scrape-potential studies). Since no classification structure has been adopted for inhibitors of SCC, various inhibitors were grouped on the basis of their chemical characteristics, i.e., cation, anion, organic, and combinations. Although several substances inhibited IGSCC in sensitized Type 304 stainless steel, $CdSO_4$ (0.5 M) was found to be the most effective inhibitor in experiments to ~600 h. The Cd^{++} and $SO_4^{=}$ acted synergistically in inhibiting crack nucleation and also arrested cracks in specimens that were initiated in uninhibited environments. This investigation, as well as other studies cited in the review,[10] indicates that the inhibitor approach shows promise; however, $CdSO_4$ addition to a BWR coolant is not a viable method because of the high neutron-capture cross section of cadmium and deposition of the nonvolatile inhibitor on the fuel cladding in regions of the core where boiling occurs.

The addition of hydrogen, ammonia, or hydrazine to high-purity water is being considered as a method to suppress radiolytic oxygen and hydrogen peroxide formation in the BWR coolant,[12-17] which will decrease the corrosion potential of the austenitic stainless steel and possibly preclude cracking of the material

under reactor operating conditions. Based upon an engineering evaluation, the rates at which these species must be added to the feedwater to decrease the oxygen concentration to desired levels or to retain measurable quantities in the core-exit water are shown in Table III.5 along with estimates of the cost.[14]

Table III.5. Consumption Rate, Cost, and Availability of Additives to the Feedwater of a 2500-MWt BWR for Suppression of the Oxygen Concentration in the Core-exit Water[14]

Species	Consumption Rate			Cost, $/day	Availability
	kg/s	kg/h	lb/day		
Hydrogen[a]	0.0026	9.35	500	700	Bulk liquid
Ammonia[b]	0.13	468	24,000	1,200	Anhydrous liquid
Hydrazine[c]	0.13	468	24,000	44,000	Not available in this amount

[a]Hydrogen addition rate would produce an oxygen concentration of ∿10 ppb in the core-exit water.

[b]Ammonia addition rate would produce an ammonia concentration of ∿10 ppm in the core-exit water, which is sufficient to prevent nitrate/nitrite formation.

[c]Owing to radiolytic and thermal decomposition, hydrazine rapidly forms ammonia and hydrogen; consequently, the rate of hydrazine addition is equivalent to that for ammonia.

These additives suppress the oxygen concentration in the reactor coolant in the following manner.

Hydrogen: The simplest scheme for the decomposition of water by radiolysis involves the formation of hydrogen peroxide ($2H_2O \xrightarrow{rad} H_2 + H_2O_2$) and free radicals ($H_2O \xrightarrow{rad} H^+ + OH^-$). Thermal decomposition of the hydrogen peroxide and recombination of the free radicals through a chain of reactions produce the molecular products H_2 and O_2. "Excess" hydrogen in solution shifts the equilibrium in favor of the reaction to form water. The steady-state concentrations of hydrogen and oxygen in the water depend on temperature, pressure, flow rate, power density, and the rate at which gases are removed from solution. During normal operation, most of the hydrogen and oxygen gases are stripped from the water and carried with the steam to the condenser where they are removed by the steam-jet air ejectors. The measured partition of the gas concentration ($H_2 + O_2$) between the steam and water at the separators (i.e., core-exit water) is 100 to 1. A distribution based upon equilibrium between the steam and water phases would yield an even higher value of the partition coefficient, which is an important parameter in relation to the amount of additive required to achieve desired levels of dissolved hydrogen in the core-exit water. Hydrogen additions at a rate of ∿4.8 and 9.3 kg/h to the feedwater of a 2500-MWt BWR would result in core-inlet hydrogen concentrations of 125 and 230 ppb, respectively, and oxygen concentrations in the core-exit water of 50 and 10 ppb, respectively, compared with a normal oxygen concentration of ∿200 ppb.[13]

Ammonia: The addition of ammonia to the feedwater provides another method to suppress oxygen production in the coolant.[12-16,18] The ammonia reacts with OH^- radicals, i.e., precursors of O_2, by the possible sequence of reactions

$$NH_3 + OH \longrightarrow NH_2 + H_2O$$
$$NH_2 + NH_2 \longrightarrow NH_3 + NH$$
$$NH + NH \longrightarrow N_2 + H_2,$$

which result in the formation of N_2 and H_2.[18] The ammonia addition rate in Table III.5 is based upon a minimum ammonia concentration of ∿10 ppm in the core-exit water to minimize the formation of nitrate and nitrite by radiolysis and maintain the oxygen concentration at ≤ 10 ppb.

Hydrazine: In principle, hydrazine can suppress the oxygen concentration by several mechanisms. At low temperatures (≤200°C), hydrazine reacts with oxygen in water to form nitrogen ($N_2H_4 + O_2 \longrightarrow N_2 + 2H_2O$); however, at higher temperatures (∿290°C) in the reactor environment, hydrazine rapidly decomposes to form ammonia, hydrogen, and nitrogen ($N_2H_4 \xrightarrow{rad} NH_3 + \frac{1}{2}H_2 + \frac{1}{2}N_2$). Because of rapid radiolytic and thermal decomposition, it is not possible to maintain trace amounts of this compound in the core-exit water. Depending on data for the thermal decomposition rate, hydrazine addition rates of ∿91,000[13] and 6,000[15] kg/h are required to obtain 8 to 25 ppm hydrazine in the core-exit water. When added at a rate of ∿470 kg/h, the ammonia concentration in the water would be sufficient to suppress the oxygen concentration to ∿10 ppb and minimize nitrate formation irrespective of the difference in the thermal-decomposition rate data.[15]

Other factors in addition to the optimum concentration, cost, and availability of various additives must be considered relative to their use in a BWR environment. For example, under normal BWR water-chemistry conditions, ^{16}N, which forms because of the ^{16}O (n,p) reaction, is quickly converted to the

nonvolatile anionic species NO_2^- or NO_3^- and only a small fraction enters the steam phase although this is the primary source of radioactivity in the steam. As the oxidizing potential of the coolant is reduced by additives such as H_2, NH_3, or N_2H_4, the proportion of ^{16}N that is converted to the relatively volatile NH_4^+ cationic species increases markedly. Thus, the amount of radioactivity released to the steam increases and could influence shielding requirements in the turbine building and contribute to the dose rate of personnel.* Furthermore, each of the additives produces volatile decomposition products that are continuously stripped into the steam phase at the top of the reactor vessel, extracted from the condenser, and passed to the off-gas treatment system. The size of the components and piping in present BWR systems may or may not be adequate, depending upon the additional gas flow due to the additives. Also, the change in composition of the off-gas could alter the performance of the recombiner.[14]

The use of ammonia or hydrazine would also necessitate alternate methods for the detection of condenser leakage, since these additives increase the conductivity of the condensate to a level that decreases the sensitivity of the measurement significantly. Reliable ion-selective electrodes (e.g., Na^+ or Cl^-) with a sensitivity of $\sim$0.1 ppb would be required to monitor condenser leakage. Other considerations relative to the use of volatile additives include demineralizer performance, optimum locations for additive injection and monitoring, special safety and toxicity requirements, feasibility and economics of hydrogen separation and recycle, and the influence of additives on the structural integrity of system materials.[15]

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*Steam-line radiation levels have been found to increase by factors of 2 to 10 during hydrogen additions to the coolant of BWRs and test facilities that simulate BWR conditions.[15]

The possibility that an alternate water chemistry can lead to a materials-coolant incompatibility within the system must be carefully evaluated. The influence of hydrogen and ammonia additions to the feedwater on the chemistry of steam entering the turbine is shown in Table III.6 along with information for typical BWR and fossil-fired systems.[14]

Table III.6. Comparison of Turbine Inlet Steam Chemistry for Typical BWR and Fossil-fired Plants with BWR Plants Based on Alternate Water Chemistries (H_2 or NH_3 Additions)[14]

System	Oxygen Conc.	Hydrogen Conc.	Ammonia Conc.	pH
Typical BWR	20-30 ppm	2-4 ppm	-	7.2
H_2 addition to BWR	1 ppm	2 ppm	-	7.2
NH_3 addition to BWR	-	14 ppm	37 ppm	9.5-10.5
Typical fossil-fired plant	<10 ppb	<10 ppb	<1 ppm	8.5-9.5

Relative to a typical BWR, ammonia added to the feedwater increases the pH and the concentrations of hydrogen and ammonia in the steam considerably, whereas hydrogen additions only decrease the oxygen concentration of the inlet steam to the turbine. An alternate water chemistry based upon ammonia

additions to the coolant would also require a careful assessment of the susceptibility of steels in the turbine to hydrogen embrittlement, and the potential for cracking of the brass steam packings. These factors, coupled with the fact that hydrogen is the only additive that will not degrade the performance of the reactor water treatment systems in the event of condenser leakage*, virtually preclude alternate water chemistries based on ammonia or hydrazine additions to the coolant. The precedent for the use of hydrogen to suppress radiolytic decomposition of water is well established in the primary heat-transport system of PWRs where the incidence of IGSCC has not been a major problem.

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*Ammonia and hydrazine have a highly deleterious effect on the ion-exchange capacity, efficiency, and regeneration requirements for both the anion and cation resins in deep-bed ion exchangers. Ammonia carried over in the steam is converted to ammonium hydroxide in the condensate and produces a high loading of the condensate and water clean-up streams. Thus, only a small fraction of the capacity of the resins would be available to control the concentrations of the Na^+ and Cl^- ions in the event of condenser leakage in a sea- or brackish-water-cooled plant. Ammonia additives also would increase the input to the radwaste system and the volume of stored waste to impracticable levels.[14,15]

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F. Basic Understanding of SCC

Stress-corrosion cracking is conventionally considered to be the failure (with limited deformation) of a material in a particular "susceptible" condition, in a particular environment, and at a particular temperature, due to the "initiation" and propagation of a sharp crack when the material is subjected to a fixed tensile load which, in the presence of an "inert" environment, or for a "non-susceptible" material condition, would lead to no failure or a "ductile" failure with large deformation. Although this long list of "fixed" or "particular" conditions is motivated far more by considerations of convenience in laboratory testing than by a recognition of realistic operating conditions of real engineering components, this restricted use of the term SCC has strongly permeated the design philosophy for real situations.

This can be illustrated by considering a schematic representation of the type of data often used in design, as shown in Fig. III.1. A series of "smooth" test specimens are exposed to the environment and temperature of interest at different fixed levels of stress and the times to failure are determined. Sometimes results are obtained in an "inert" environment as well as in the "active" environment of interest, and of course the general effect of the "active" environment (solid curve in Fig. III.1) is to shorten the times to failure compared with those in the "inert" environment (dashed curve).

A key parameter often reported from such studies is a "threshold" stress (σ_{th}) below which it is presumed that SCC will not occur. The desire of the designer for a "material property" such as σ_{th} is quite understandable, for he then has the prospect of designing his structure so that σ_{th} is not exceeded and hence SCC is avoided. Unfortunately, an almost unlimited list of difficulties

may be encountered with such an approach. The failure curve shifts with thermo-mechanical treatment of the specimen, stress state, environment, temperature, etc. Assuming that all these variations could be adequately explored through a sufficiently generous test matrix and accounted for through a reasonable analysis tool, there remain some additional and especially troublesome factors.

One concerns the meaning of σ_{th}. It is extremely difficult, if not impossible, to prove that no failure will occur below σ_{th} no matter how long one waits. Thus, σ_{th} normally is, in fact, a measure of the stress required for failure in a certain time t_{th}, assuming, of course, that the applied load remains fixed. Rare indeed is the application for which laboratory tests can be run in representative environments at representative temperatures for the design lifetime of the component, so extrapolations in time are the rule. Allowing for all this may, more often than not, make it impossible to design such that the "threshold" for SCC can be confidently avoided, especially when one considers all contributions to loading (residual stresses, thermal stresses, etc.) in addition to "applied" loads. One is then forced to employ some type of history-dependent failure-prediction approach, and some degree of mechanistic understanding is essential for such a complex phenomenon.

Another problem with the "σ_{th} approach" is the arbitrary restriction to fixed-load tests, when in fact applications rarely even closely approximate such a condition. If the load variations are rather "gradual", one might hope to do a sufficiently accurate time integration based on fixed-load results; but more difficult questions arise when one considers the real possibility of oscillating or cyclic loading superimposed upon more slowly varying average loads. Here again, oversimplifications based on convenience in laboratory testing procedures have introduced considerable confusion.

Since failure under cyclic loading is conventionally labeled as fatigue, recognition of the effect of cyclic loads on environmental cracking led to the definition of a new term, "corrosion fatigue" (CF), which is normally treated as a separate design problem, with its own rules and procedures closely based on the particular way the laboratory tests are (conveniently) run. In fact, at least one group of investigators[1] has proposed the term "true corrosion fatigue" when the experimental results exhibit certain arbitrary characteristics and the term "stress corrosion fatigue" when they do not. We strongly agree with Ford, who says[2] that such "rigid categorizations are questionable ... artificial and confining". We further agree that the categories of SCC and CF can be very confining and artificial, and that a mechanistically based modeling of these areas would be much more profitable. Ford and his co-workers have obtained some promising results in one such integrated approach.[2] In particular, for the case of sensitized austenitic stainless steels in LWR environments, which is of principal concern to us here, they have shown that σ_{th} is in no way a material parameter when any significant cyclic loading is applied, since cracking can then occur at mean stresses far below the nominal σ_{th} obtained under constant loads (reportedly, roughly equal to the yield strength). Parkins[3] reports similar results for other systems and considers it a result to be expected generally. Indeed, recent work by Takaku et al.[4] with sensitized Type 304 stainless steel at 290°C in high-purity water with ∿8 ppm oxygen showed that IGSCC is also facilitated by cyclic loading. This work also further demonstrated the key role of strain rate in constant extension-rate testing, since susceptibility increased with decreasing strain rate to 4.2×10^{-7} s^{-1}, the lowest employed. Berry et al.[5] have also reported that cyclic loads can

produce IGSCC under similar conditions, at applied stresses at least as low as half the nominal yield strength. The authors again emphasized the key role of low strain rates in promoting IGSCC. In this vein, results by Vyas[6] indicate a maximum crack growth rate at strain rates $\sim 10^{-6}$ s^{-1}, which agrees very well with theoretical predictions by Parkins.[3]

Another problem with the σ_{th} approach is the effect of surface condition and particularly the existence of "flaws" or other "stress concentrators". It is difficult if not impossible to guarantee that a real structure will contain no "flaws" worse than the ones existing on the "smooth" laboratory test specimens used to establish the time-to-failure curves. This understanding, coupled with the great popularity of LEFM among designers, inevitably led to the adoption of LEFM techniques for assessing crack growth under fixed loads (SCC) and cyclic loads (CF), with insufficient attention paid to the validity of such an approach. Again, standardized testing procedures dictate the design rules. Illustrations of data presentations are given in Figs. 2 and 3 for fixed load and fixed cyclic amplitude, respectively. Both approaches are, of course, borrowed from the previously developed approaches for crack growth in an "inert" environment.

Figure 2 displays the time rate of growth of crack size, a, as a function of the applied stress intensity K. The general shape of the resulting curve is typically "sigmoidal", and "power-law" fits are often made to the regions marked I, II and III. In "inert" environments the power-law exponent n in the expression

$$\frac{da}{dt} = CK^n , \qquad (1)$$

where C is a constant, is sometimes reported to be large; but this can really be considered only a very crude estimate of an "average" value. For "active" environments, n is typically smaller in region II, and one often sees plots like that of Fig. III.2, with stage II drawn horizontally (n = 0), referred to as "typical" of SCC. Again, this must be considered an oversimplification; surely there is a continuous spectrum of environmental severity from "inert" to highly aggressive and it would be extremely naive to expect a sudden change from a high value of n to n = 0 when such a spectrum is traversed. In spite of this, n = 0 is so often quoted as "typical" for stage II that one often sees SCC design approaches with n = 0 assumed to hold true over the entire range of application. The same discussion and the same reservations and cautions apply to stage II of Fig. 3, except that here one writes

$$\frac{da}{dN} = C'(\Delta K)^n, \qquad (2)$$

where N = cycles, C' = constant, and $\Delta K = K_{max} - K_{min}$. A value often assumed for n is 4; another commonly assumed value is 2, but again a wide range of values is found experimentally.

The critical values at low K and ΔK are similar to σ_{th} in that they are often assumed to be material parameters, whereas in fact they are often simply the values at the limit of sensitivity for measurement of crack advance under very specific conditions. Also, studies by Wei[7] raise very serious questions about the meaning of K_{ISCC}, since he found its value to depend upon the initial K-value employed in a particular test.

The critical values at the upper end of stage III are often arbitrarily assumed to correspond to K_{IC}, the so-called plane strain fracture toughness

value for unstable (fast) fracture although quantitative data supporting this assumption are, like those supporting n = 0 in stage II, often lacking. Also, specimens employed typically fail to meet "plane-strain" validity criteria.

It has been increasingly recognized that the plot in Fig. III.3 is affected not only by the range ΔK but also by the absolute levels of K employed. The parameter used to describe this effect is usually $R \equiv K_{min}/K_{max}$ or, equivalently for a given specimen/crack geometry, $\sigma_{min}/\sigma_{max}$. Wide variations with frequency are also often seen.

Although it is often stated or presumed that crack "initiation" occupies a major portion of the t_f values in Fig. III.1, the LEFM approaches embodied in Figs. III.2 and III.3 are typically applied to structures with an assumed flaw size and shape at time zero. This is another inconsistency between the two types of approaches, though the difficulty, if not impossibility, of guaranteeing a flaw-free structure probably makes the LEFM approach more realistic for safety design.

Whether or not the entire LEFM framework is applicable to SCC, with its implicitly assumed independence of stress state, is a question which requires much more careful scrutiny than it usually receives. As a minimum, SCC studies should employ at least two types of specimen which differ widely in their type of loading; in contrast, many studies employ only one and simply assume the validity of the LEFM K-characterization. If such studies do not produce results that are independent of specimen type, much more basic or mechanistically oriented studies will be required to develop reasonable design procedures. As has been pointed out elsewhere,[8] this need is not peculiar to SCC. In this vein, comparative studies of Mode I and II loadings[9] would be of great interest for both mechanistic and predictive interests.

The overly simplified and sometimes contradictory special cases of design approaches discussed above point out the great need for a unified treatment of crack propagation in "active" environments which can somehow cope with realistic load/temperature/environment histories anticipated in the service of an actual component. This is an extremely difficult task and should be approached only in carefully selected "pieces" for which some hope of success exists. In the present context, Type 304 stainless steel in LWR environments is of particular interest and some progress has been made.

Ford and co-workers[2] have examined the continuum in the behavior of sensitized Type 304 stainless steel in a BWR environment under "static" to fully reversed "cyclic" loading and have developed a unified treatment which offers the potential of a predictive tool for an "arbitrary" load history. They first mesh the "smooth-specimen" and "fracture-mechanics" approaches, as illustrated schematically in Fig. III.4, where the left vertical axis represents the "smooth"-specimen approach and the curves, varying with crack depth, illustrate the effect of flaws or cracks. The K_{IC} curve is for an "inert" environment and the K_{ISCC} curve is for the environment of interest. They then account for the fact that this latter curve is affected by cyclic stresses superimposed upon the average "static" stress, and in this manner mesh static and cyclic loading conditions. The intermediate region is assessed by crack-growth relations, modified for the effects of R, cyclic frequency, and precise shape of the load vs time curve. This approach has been developed from a mechanistically oriented research program and, although only semi-quantitatively developed to date, illustrates the potential of such an approach. It has the advantage of building upon a familiar, i.e.,

fracture-mechanics-type approach.

The type of mechanistic approach envisioned is one which employs mechanistic or basic concepts to develop unified analytical approaches which obliterate the arbitrary and often misleading categorizations of SCC, CF, etc. The eventual aim is a fully predictive tool for the behavior of Type 304 stainless steel as a function of microstructure, particularly with varying degrees of cold work and sensitization, in LWR environments and under realistic load/temperature/environment histories. This, however, must be understood as a long-term goal; even before its attainment, a mechanistically oriented approach should prove extremely valuable in interpreting and applying experimental data wisely and in devising more appropriate and useful experiments.

The Ford approach is closest in spirit to the proposed studies, and it will be examined in detail for at least partial implementation into this effort. In addition, other basic mechanistic efforts will be carefully studied to draw from them the insight needed for a reasonable intuitive approach that concentrates on general concepts, in contrast to the more academic approaches that focus on detailed "mechanisms" such as "strain-assisted active path" or "hydrogen embrittlement". However, the most fruitful area for concentration promises to be "rate-controlling" sub-processes which may well be common to the various specific mechanisms for crack advancement. This approach has already produced results[2] that strongly question the view expressed in a recent review:[10] "Theoretical studies are required for understanding for the alloy development, but information on another scale is required for practical control measures".

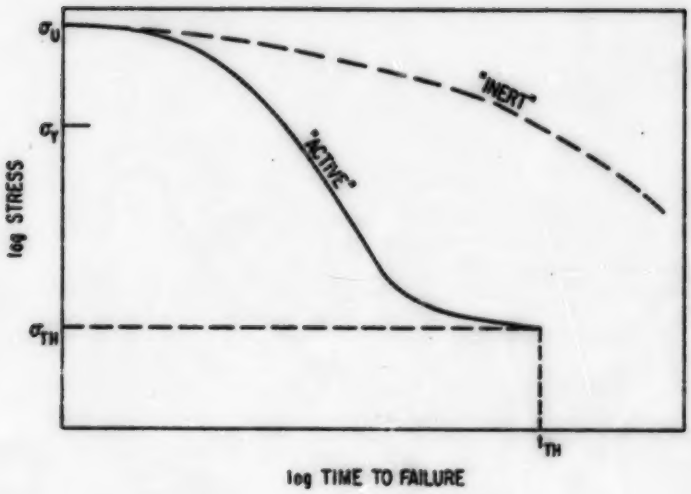

Fig. III.1. Schematic of Time to Failure under Static Loads.

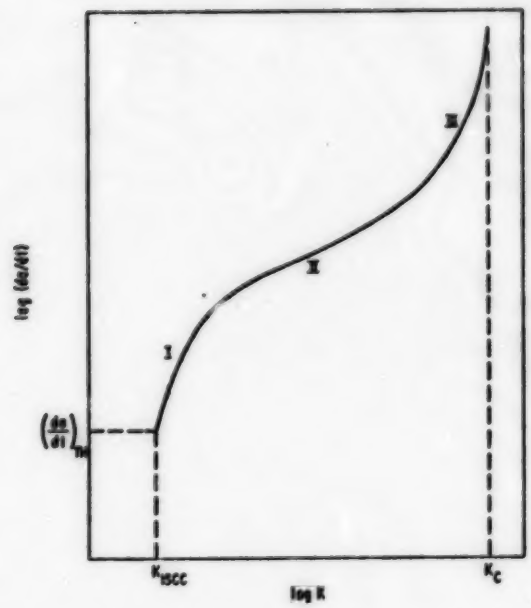

Fig. III.2. Schematic of Crack Growth Rate as Function of Static Stress Intensity.

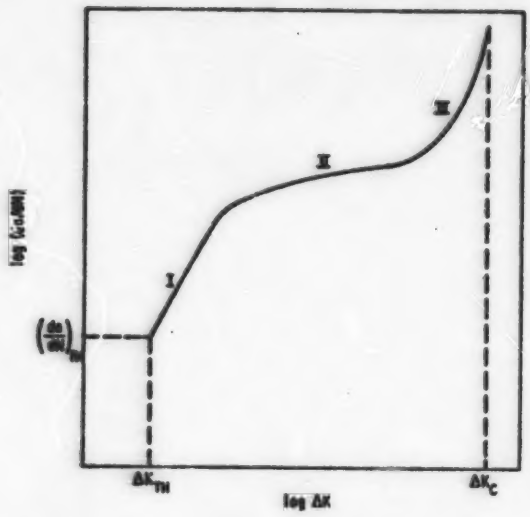

Fig. III.3. Schematic of Crack Growth Rate as Function of Cyclic Stress Intensity Range.

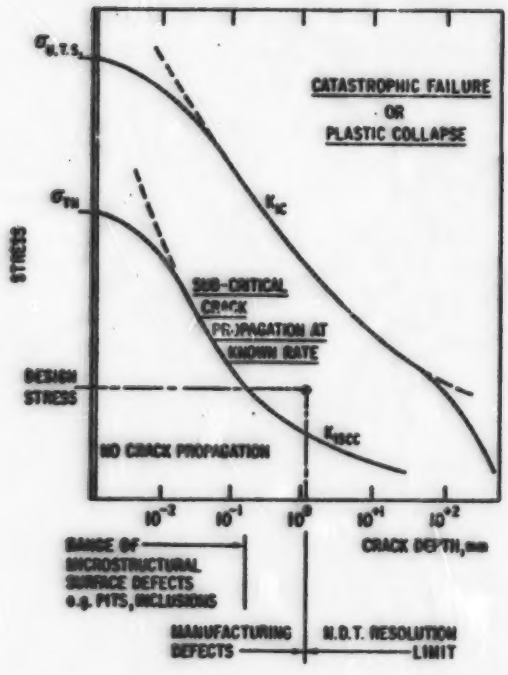

Fig. III.4. Tensile-stress/Crack-depth Diagram, Illustrating Regions of "No Crack Growth", "Subcritical Crack Growth" and "Catastrophic Failure" (or Plastic Collapse).

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5. W. E. Berry, E. L. White and J. H. Payer, "Stress Corrosion Cracking of Sensitized Type 304 Stainless Steel under Cyclic Load," ibid., Vol. 2, Paper 30.

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8. F. A. Nichols, J. Eng. Fract. Mech. 12, 319 (1979).

9. Anthony W. Thompson, Mater. Sci. and Eng. 43, 41 (1980).

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IV. ASSESSMENT OF RESEARCH AND DEVELOPMENT NEEDS

Based on the foregoing discussion of the technical issues and the ongoing utility/industry efforts on SCC problems, the following research and development needs have been identified as essential to the development of an independent capability for evaluating SCC problems and their potential solutions. The logic for a program addressing these needs is outlined in charts at the end of each section. The technical goals are listed on the charts in a rough ranking of priority and time sequence. The relationship of the technical goals to the overall program goals is indicated in Table IV.1. It is estimated that a five-year period will be required to complete the program,* assuming that adequate funding is available and that the problems do not continue to escalate.

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*Except for ASME Code modifications.

Table IV.1. Relationship of Technical Goals to Overall Program Goals

Research Area	Action Needed to Define Problem	Evaluation of Proposed Solutions	Actions Resulting in Improved Margin: Existing Plants	Actions Resulting in Improved Margin: New Plants	NRC Goal	Ultimate Goal
Leak Detection and NDE	Evaluate State of the Art.	Quantify Improved Methods in Lab.	Demonstrate on Plant Components (Lab & Field).	Demonstrate for New Materials.	Input for Reg. Guide.	Revise ASME Section XI.
Analysis of Sensitization	Quantify LTS Kinetics.	Are LTS Test Results Conservative?	Use "LTS" in Testing.	Quantify LTS for Alt. Materials.	Input for Reg. Guide on Qualifying Tests	
	Define Limits of EPR Technique.	Determine reliability in Field Applications.	Use EPR to Qualify Alt. Materials.	Use EPR to Qualify Alt. Materials.		
Analysis of Crack Growth	Define Margin for Cracked Pipe.	Determine Validity of LEFM for IGSCC.	Combine with Residual Stress to Predict Lifetime in SS.	Obtain Data on Alt. Materials.	Input for Reg. Guide on Procedures for Cracks in Large Pipes.	
Nonenvironmental Corrective Actions	Review Stress "Remedies"	Quantify Margin Improvement.	Use in Repairs.	Use in New Construction.	Input for Reg. Guide on Materials and Fabrication Procedures.	Revise ASME Section III.
	Review Alt. Materials.	Evaluate Possible Side Effects.	Use in Repairs.	Use in New Construction.		
	Full-scale Pipe Testing.	Check Basis and Validity of Test Procedures.	Determine Quantitative vs. Qualitative Nature of Margins. Devise Supplemental Tests.		Input for In-Plant Tests.	
Environmental Corrective Actions	Evaluate Oxygen and Impurity Control.	Conduct Laboratory Tests.	Perform Field Demonstration; Evaluate Side Effects; Establish Tighter Water Specs.	Evaluate Effects on Alt. Materials.	Input for Reg. Guide on Oxygen and Impurity Control.	
	Evaluate Need for LWR Loop Tests.	Perform In-Plant Measurements.	Compare Field and Lab. Chem. Data.	Install Plant Loops.	Validate Lab Testing.	
Basic SCC Understanding	Evaluate Current Theory.	Validate Via Lab Tests as Required.	Develop Improved Model for SS.	Extend Model to Alt. Materials.	Use Model to Verify "Solutions" to SCC by Extrapolation.	

A. Leak Detection and Nondestructive Evaluation

1. Leak Detection

A more effective system for detecting small leaks is required to meet the concerns and needs of the NRC. This is particularly evident when it is realized that relatively small leaks may occur even with extensive cracking, as the Duane Arnold safe-end failure made clear. Current leak-detection techniques are not capable of simultaneously providing adequate detection sensitivity, measurement accuracy, and location capability.

Acoustic emission techniques seem potentially capable of satisfying these requirements, but a considerable amount of effort is required before an acoustic surveillance system can be implemented effectively. An assessment of the capability of acoustic emission technology to detect leaks in both BWRs and PWRs should be a high-priority research and development effort. The ability of acoustic monitoring techniques to measure leak rates and locate naturally occurring cracks in piping under field conditions needs to be established. The effects of crack geometry on leak rate and associated acoustic signal generation must be determined.

Initial studies can be carried out in a laboratory test facility in which crack-like leakage under simulated reactor conditions at high and low pressures can be studied. The system should operate at reactor temperatures (up to 600°F) and pressures (up to 2000 psi), and be designed so that various pipe sections containing natural and artificial leaks can be introduced into the piping, including decontaminated (but still possibly radioactive) piping with field-induced throughwall cracks. The ability to study leak signals from field-induced throughwall cracks (especially as a function of distance) is particularly important, as this type of work has not yet been carried out. Background noise signals from operating reactors can be obtained and then compared with the leak signals from the test facility.

Acoustic emission sensors with bandwidths up to 1 MHz appear to be the most suitable for leak-detection systems. Frequency spectra must be examined to determine acoustic leak signatures as a function of pressure, temperature, and flow rate, and the use of cross-correlation techniques to increase signal to noise ratios of leak signatures and locate leaking should be studied. Matched-filtering techniques could also be examined to improve signal-to-noise ratios for leak detection and pattern recognition techniques for source discrimination. Studies on minimum detectable leak rates and leak-geometry effects should be carried out. Threshold levels for detecting leaks should be estimated and leak detection and location schemes established.

The laboratory test system could not only be a valuable tool in the study of improved acoustic leak-detection systems, but could also be used to evaluate the sensitivity of existing commercially available leak-detection systems and possibly other leak detection schemes. A laboratory system could also be used for establishing calibration procedures and techniques for attaching transducers to thermally hot pipes.

The overall program outline is shown in Fig. IV.1.

2. Nondestructive Evaluation

The limitations of current ultrasonic ISI techniques suggest a need for improvement of conventional testing methods and research and development to develop new ultrasonic techniques. Conventional methods need improvement in the areas of standardization of procedures, standardization and better characterization of equipment, and personnel training. This may include, for example, requirements to assure the quality of ultrasonic probes through beam-profiling techniques and frequency analysis. There is also a need for specific guidance in the code with regard to inspection of stainless steel reactor piping and particularly with regard to detection of IGSCC. Some effort is also required

to determine the advantages and disadvantages of dual-element, focused single-element, and conventional single-element transducers for ISI. Many of these needs (but not all) are being met by programs sponsored by NRC (at PNL) and EPRI (at Southwest Research Institute, Battelle-PNL, and elsewhere). Relevant NDE programs are summarized in Appendix A.

Another important area is that of calibration standards. Access to components identical with or similar to the component to be examined and including typical flaws could be important in improving inspection reliability, operator confidence, and skill.

There is also an obvious need to develop an ultrasonic system that can effectively and rapidly discriminate between geometrical reflectors and IGSCC. Although EPRI is funding work of this type (see Appendix A), additional research in the areas of signal analysis and pattern recognition techniques for flaw detection is required. There is a need to identify signal characteristics that will allow cracks to be distinguished from geometrical and metallurgical reflectors in a reliable and practical way during field testing. Most of the various systems currently under development and evaluation are based on pattern recognition techniques applied to the analysis of data obtained by conventional UT probes. Another approach that has not yet been examined is a multi-element probe that would essentially detect the scattered ultrasonic wave pattern from a flaw (i.e., map out to some extent the scattered wave). Because of the rough surface and branching characteristics of IGSCC, the scattered wave from a crack may be different than one from a geometrical reflector. A simple tri-element probe appears to have some potential. One element would transmit a pulse and all three would receive the scattered wave. The relative signal amplitudes of the received echoes would be a quantitative measure of the energy distribution of the scattered wave pattern. Although the idea is speculative at present, an attempt to discriminate between IGSCC and geometrical reflectors through the use of scattered wave patterns would complement current efforts that emphasize signal processing.

The possible presence of cracks in Type 308 stainless steel weld metal and the necessity to examine the HAZ in a through-the-weld ultrasonic inspection configuration point to the need for development and evaluation of longitudinally and horizontally polarized shear-wave probes. Longitudinal waves can be more effective than the conventional vertically polarized shear waves in angle-beam testing because the anisotropic nature of the weld metal produces distortion and attenuation of the shear waves. The ability of longitudinal-mode angle-beam probes (including focused beams) to detect weld-metal flaws and carry out through-the-weld inspections should be assessed. A prototypical longitudinal-mode angle-beam probe for weld-metal inspection and through-the-weld inspection of the HAZ should be designed and fabricated. The adequacy of this probe should be compared with that of others available from the NDE industry, using available samples containing field-induced and laboratory-grown IGSCC and artificial reflectors. Probes that can be used successfully for stainless steel weld metal inspections may also be useful for coarse-grain cast stainless steel pipes, which have a similar anisotropic structure.

Another area that should be explored is the possibility of employing EMATs for through-the-weld metal inspections. These probes require no couplant and can be used to generate horizontally polarized shear waves. Although less efficient than conventional transducers, they could be more effective than conventional shear-wave probes in special situations, as alluded to in Section III.A.

The potential for improving NDE of reactor components through developments in phased arrays, spatial averaging techniques, and automated equipment should be assessed. Research should be accelerated in the most promising areas. Efforts in these areas are discussed in References 24, 25, 27, 29, 43, 44 and 45. The overall program outline is shown in Fig. IV.2. Appendix A summarizes the NDE programs sponsored by NRC, EPRI and DOE that involve research in these areas.

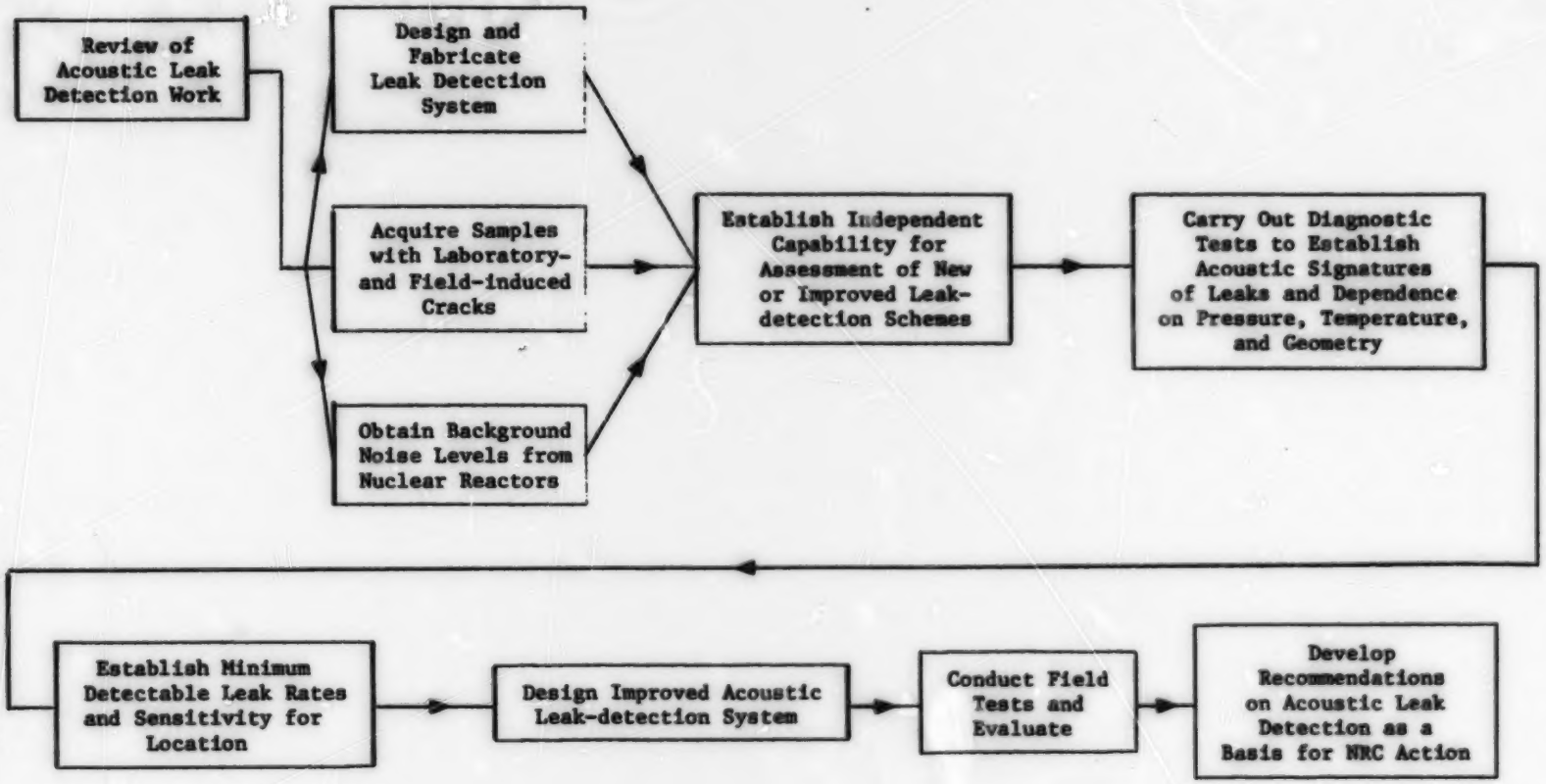

Review of Acoustic Leak Detection Work
Design and Fabricate Leak Detection System
Acquire Samples with Laboratory- and Field-induced Cracks
Obtain Background Noise Levels from Nuclear Reactors
Establish Independent Capability for Assessment of New or Improved Leak-detection Schemes
Carry Out Diagnostic Tests to Establish Acoustic Signatures of Leaks and Dependence on Pressure, Temperature, and Geometry
Establish Minimum Detectable Leak Rates and Sensitivity for Location
Design Improved Acoustic Leak-detection System
Conduct Field Tests and Evaluate
Develop Recommendations on Acoustic Leak Detection as a Basis for NRC Action

Fig. IV.1. Outline of Proposed Program on Acoustic Leak Detection

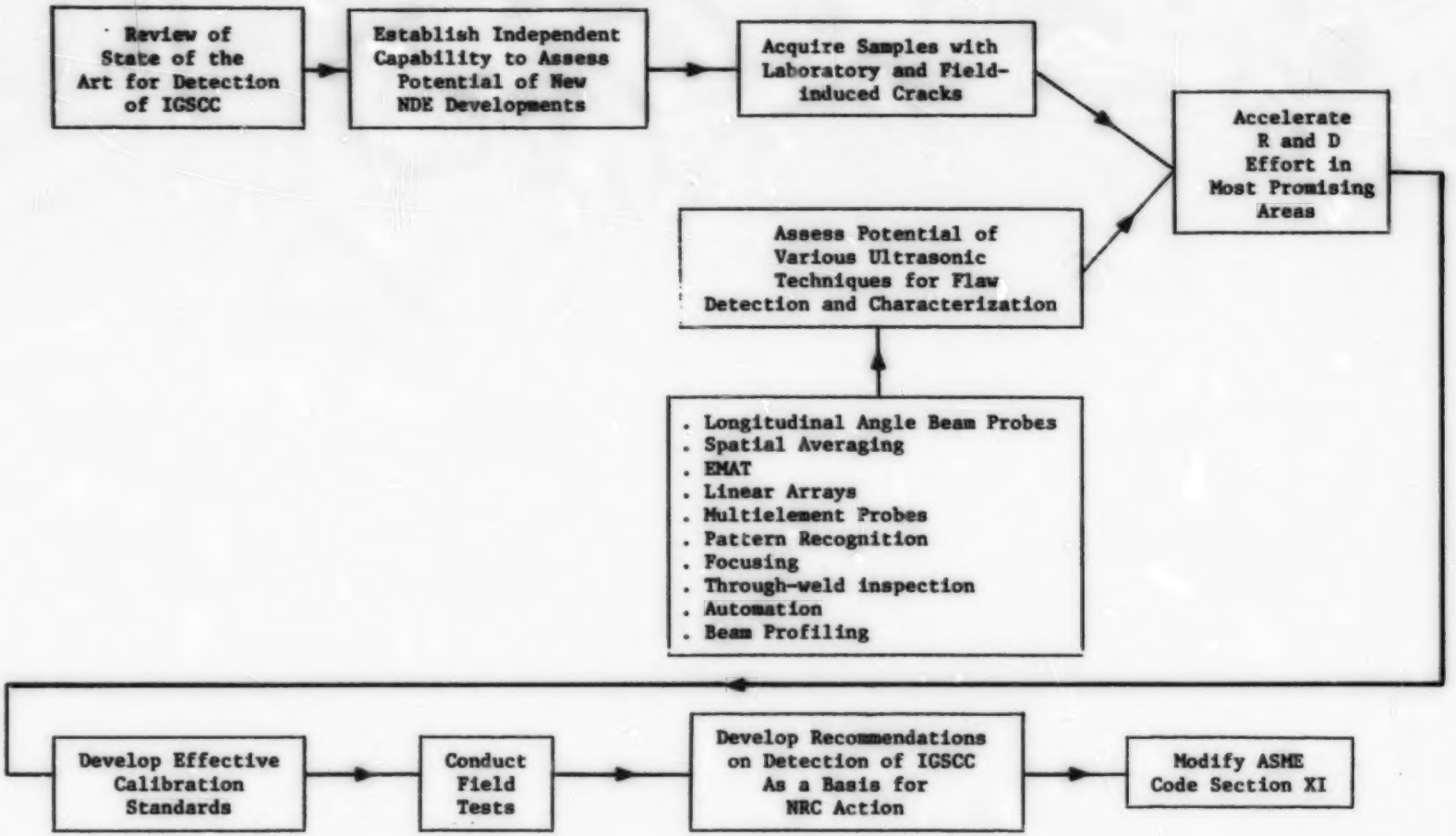

Review of State of the Art for Detection of IGSCC
Establish Independent Capability to Assess Potential of New NDE Developments
Acquire Samples with Laboratory and Field-induced Cracks
Accelerate R and D Effort in Most Promising Areas
Assess Potential of Various Ultrasonic Techniques for Flaw Detection and Characterization
. Longitudinal Angle Beam Probes
. Spatial Averaging
. EMAT
. Linear Arrays
. Multielement Probes
. Pattern Recognition
. Focusing
. Through-weld inspection
. Automation
. Beam Profiling
Develop Effective Calibration Standards
Conduct Field Tests
Develop Recommendations on Detection of IGSCC As a Basis for NRC Action
Modify ASME Code Section XI

Fig. IV.2. Outline of Proposed NDE Program.

B. Analysis of Sensitization

As discussed in Section III-B, the low-temperature sensitization (LTS) phenomenon in Type 304 stainless steel in the temperature range of 350-500°C has been demonstrated in laboratory experiments. Extrapolation of these results indicates that Type 304 stainless steel BWR pipe weldments could become sensitized and thus susceptible to IGSCC within reactor lifetimes, and hence a more detailed study of the LTS phenomenon is required.

Currently, information on the kinetics of LTS is available only for a few heats of material. Laboratory experiments show large variations (40-65 kcal/mol) in the apparent activation energy for different heats of material and types of specimen. These variations are extremely important in assessing the possibility of LTS in operating reactors, since extrapolation predicts that LTS at 300°C would occur within 10 years at 40 kcal/mol and about 1000 years at 65 kcal/mole. A large heat-to-heat variation in the critical cooling rate for carbide nucleation is also noted. This heat-to-heat variation has been attributed to differences in the amount of strain, dislocation density, and/or impurity-element content of the material, but with the current level of understanding the susceptibility to LTS of arbitrary heats of material cannot be assessed. Multi-heat studies on LTS kinetics, including the nucleation stage and influences of other variables such as strain, are needed before the real importance of LTS can be determined. The possibility of using LTS heat treatments as an accelerated screening test for susceptible heats of material is also an area worthy of investigation.

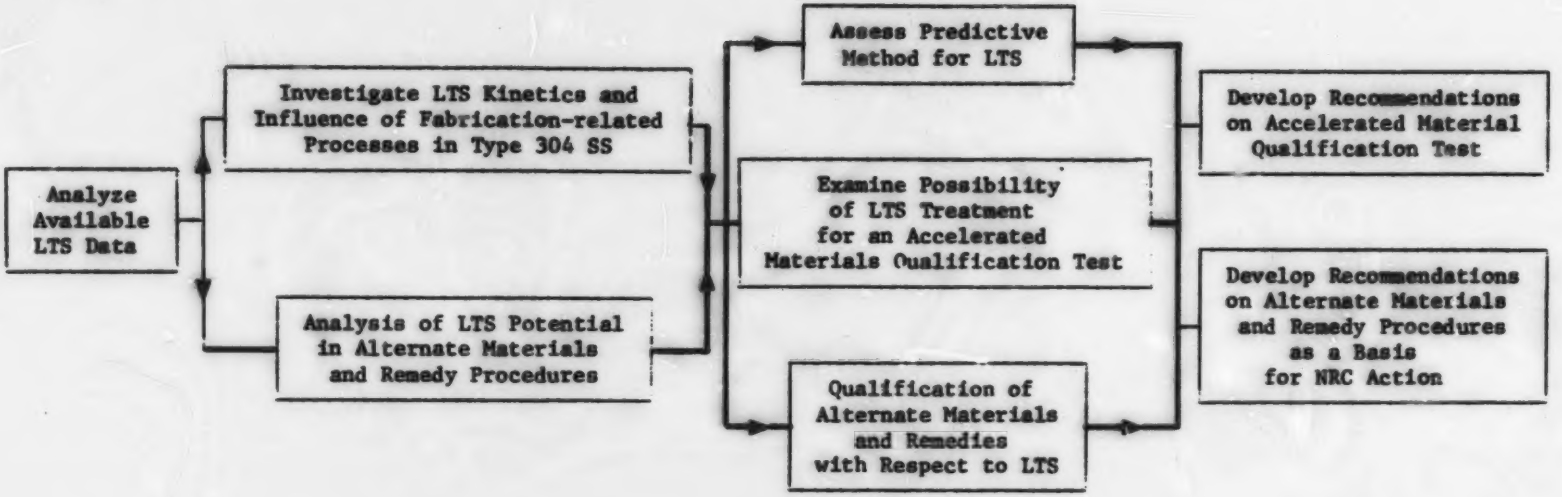
The possibility that remedial thermomechanical treatments, i.e., CRC, IHSI, and HSW, increase susceptibility to LTS also needs to be

investigated. LTS seems a remote possibility in the "shop CRC" procedure because the piping is solution heat treated after deposition of the cladding. However, the potential for LTS exists in the "field CRC" procedure, in which the cladding is deposited without SHT so that a sensitized zone remains on the inner surface of the pipe. Experimental data indicate that HSW does substantially lower the DOS for a given heat of material compared with conventional welding practice, but measurements on a susceptible heat of Type 304 stainless steel weldments prepared by HSW showed evidence of sensitization even higher than the threshold value for IGSCC, and a potential for LTS clearly exists. In the IHSI process, the temperature at the inner surface of the piping remains below 200°C and thus no effect on short-term sensitization is expected. However, the strain induced during the induction-heating cycle may influence long-term LTS kinetics.

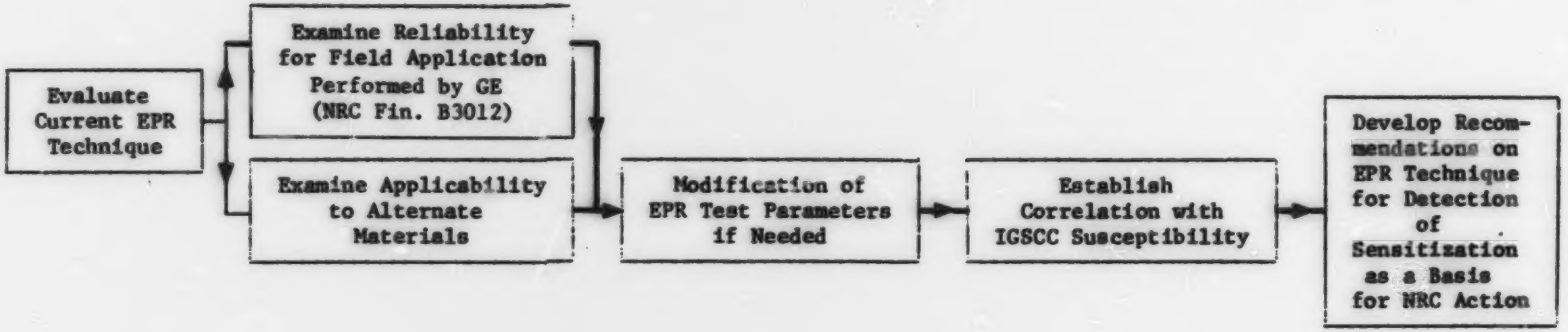
The potential for LTS in the proposed alternate materials (the "Nuclear Grade" Types 304 and 316 stainless steels and perhaps duplex stainless steels) needs to be investigated to qualify these materials for long-term service. Sensitization is suppressed in these materials; however, as long as some grain boundary carbide nuclei are precipitated and remain in the material, the possibility of LTS does exist. In the "Nuclear Grade" materials, controlled amounts of nitrogen (0.06-0.10%) are required. Some studies of the effects of nitrogen on sensitization are available, but the effects on LTS are not known at present.

The EPR technique for the determination of the DOS is relatively well established for Types 304 and 316 stainless steels. Threshold DOS values and correlations with IGSCC susceptibility have been established. However,

anomalous results have been reported on large grain size materials, and the choice of a normalization method for grain size needs to be resolved. The currently proposed normalization method does not take into account variations in the chromium concentration profile in the region adjacent to the grain boundaries. Also, since most of the EPR data base has been generated by General Electric and only a very limited amount by other laboratories, the round-robin test between different laboratories initiated by ASTM needs to be completed. It should also be noted that in some ways, ASTM A262 Practice E is a superior test method for DOS measurement; the main shortcoming is the lack of sensitivity in the low range of sensitization levels. The applicability of the EPR technique to alternative materials needs to be investigated. A correlation between the DOS measured by EPR and IGSCC susceptibility needs to be established and threshold EPR values for alternative alloys should be determined if, in fact, they exist. The overall program outline is shown in Fig. IV.3.


Analyze Available LTS Data
Investigate LTS Kinetics and Influence of Fabrication-related Processes in Type 304 SS
Analysis of LTS Potential in Alternate Materials and Remedy Procedures
Assess Predictive Method for LTS
Examine Possibility of LTS Treatment for an Accelerated Materials Qualification Test
Qualification of Alternate Materials and Remedies with Respect to LTS
Develop Recommendations on Accelerated Material Qualification Test
Develop Recommendations on Alternate Materials and Remedy Procedures as a Basis for NRC Action

(a)


Evaluate Current EPR Technique
Examine Reliability for Field Application Performed by GE (NRC Fin. B3012)
Examine Applicability to Alternate Materials
Modification of EPR Test Parameters if Needed
Establish Correlation with IGSCC Susceptibility
Develop Recommendations on EPR Technique for Detection of Sensitization as a Basis for NRC Action

(b)

Fig. IV.3. Outline of Proposed Program to Analyze Sensitization. (a) Study of low-temperature sensitization; (b) study of EPR technique for detection of sensitization.

C. Analysis of Crack Growth

The early instances of IGSCC in operating BWRs generally occurred in smaller pipes, and the response to the detection of IGSCC was generally to repair or replace the cracked piping immediately. It is now clear that for reactors with standard Type 304 stainless steel piping material, cracking can occur anywhere in the recirculation system, including the main recirculation line. Because of the severe economic consequences of long forced outages for repair or replacement, the utilities will consider other approaches for dealing with cracked pipe. The possibilities include continued operation and monitoring for any subsequent growth for an indefinite period, continued operation and monitoring until a repair can be scheduled to minimize outage, and immediate repair and replacement. Clearly, continued operation of a reactor in which cracks are known to exist must be based on a demonstration of safety despite the presence of a flaw. A credible safety analysis will require the development of reliable NDE methods to detect and monitor crack growth, the development and verification of analytical methods for predicting the remaining lifetime of flawed piping and leak-before-break behavior, and an assessment of the consequences of coolant loss for reactor operation. An assessment of research needs in the area of NDE is given in Section IV-A; assessments of needs in the areas of elastic-plastic fracture mechanics (for the leak-before-break analysis) and systems studies on coolant loss are beyond the scope of this report.

Understanding crack-growth behavior is, of course, important for other reasons besides assessing the safety implications of flawed piping. A better understanding would permit a more rational extrapolation of laboratory test results to the prediction of operating-plant behavior.

As noted in Section III.C, current estimates of crack growth rates for IGSCC in BWR environments range from 0.1 to 8 in./yr; the possible strategies available after a crack is discovered are obviously strongly dependent on which of these values is "correct". Some of this variation is undoubtedly due to heat-to-heat variations, but it also seems possible that contrary to the LEFM model, in which the crack velocity depends only on the current value of the crack-tip stress intensity K, the crack velocity strongly depends on the geometry of loading and on the nature of the remote external loading (constant extension rate, constant applied force, constant displacement).

In the long term, better predictions of crack growth will require a mechanistic understanding of IGSCC; this approach is discussed in more detail in Section IV.F. In the near term, a more phenomenological approach must be relied on, although much of the work in support of an improved phenomenological approach will be useful in mechanistic studies and vice versa. Some current mechanistic models of IGSCC in sensitized stainless steels focus on the strain rate as the critical parameter relating mechanical loading to IGSCC. Since it is difficult to measure and control strain rate in precracked specimens, initial studies should be carried out on smooth specimens. Tests should be performed with controlled strain rates and with other types of loading histories such as constant load and simple cyclic histories. Additional studies on a single heat of material utilizing precracked specimens are needed to assess the capability of K to correlate crack growth for radically different geometries and different types of loading conditions. Multi-heat studies are also needed to determine the range of propagation rates expected for the materials in operating BWRs.

Future efforts must be assessed in terms of the results obtained in the next year from EPRI- and NRC-sponsored programs. If LEFM does not provide an adequate description of IGSCC growth, new approaches must be sought. One possibility is direct characterization of strain rates at the crack tip; this clearly would present a formidable computational challenge in applications to a specific problem, and ongoing mechanistic studies should be reviewed to obtain guidance for alternative approaches. The overall program outline is shown in Fig. IV.4.

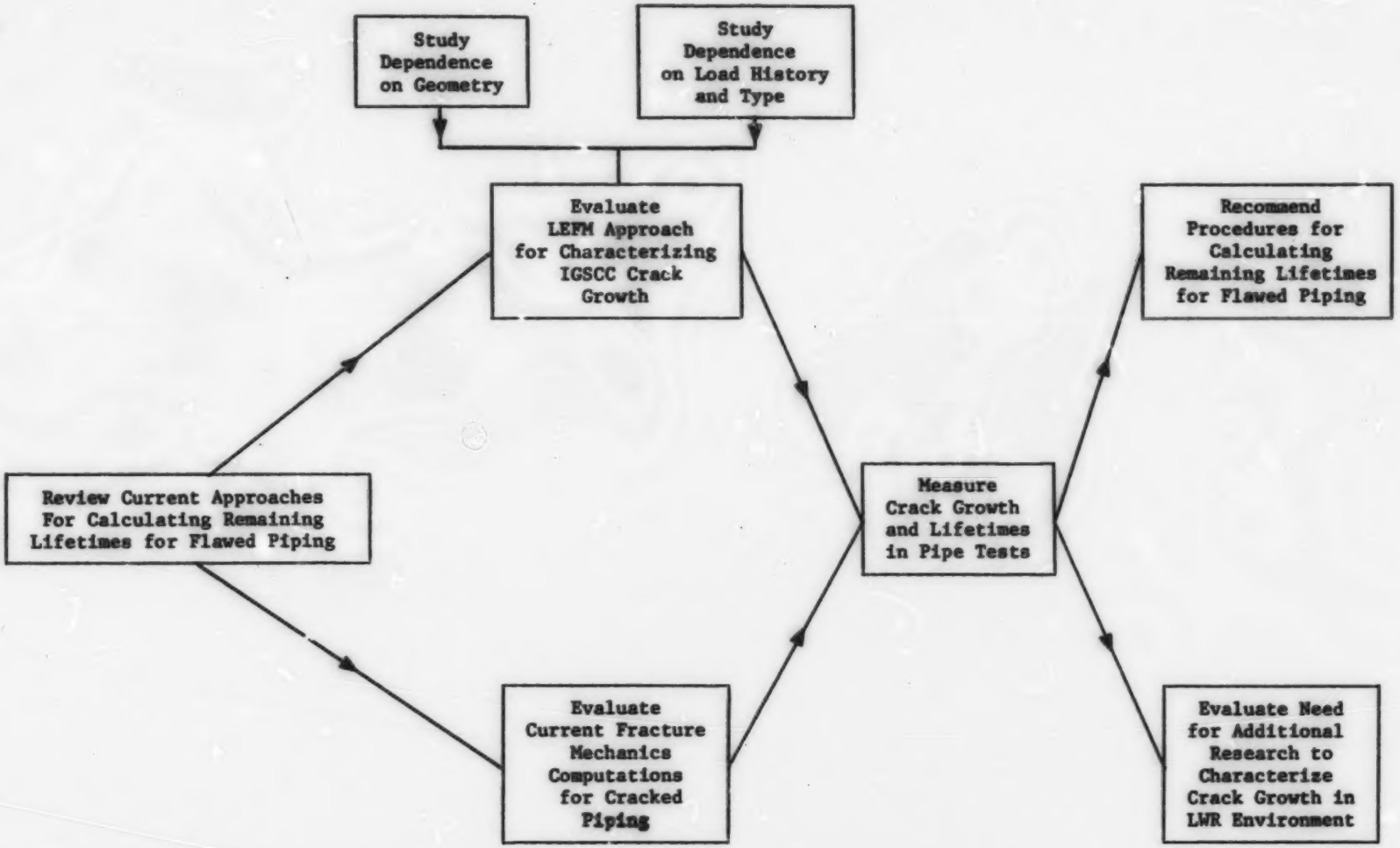

Study Dependence on Geometry
Study Dependence on Load History and Type
Evaluate LEFM Approach for Characterizing IGSCC Crack Growth
Recommend Procedures for Calculating Remaining Lifetimes for Flawed Piping
Review Current Approaches For Calculating Remaining Lifetimes for Flawed Piping
Measure Crack Growth and Lifetimes in Pipe Tests
Evaluate Current Fracture Mechanics Computations for Cracked Piping
Evaluate Need for Additional Research to Characterize Crack Growth in LWR Environment

Fig. IV.4. Outline of Proposed Program to Analyze Crack Growth.

D. Nonenvironmental Corrective Actions

The immediate problem to be addressed is corrective action for SCC in BWRs. The development of these corrective actions is primarily the responsibility of the vendors, utilities, and associated organizations such as EPRI. The qualification of the proposed remedies for this problem has been carried out mainly with EPRI support and relies heavily on a pipe testing program being carried out at GE. Thus, not only must the remedies themselves be evaluated, but the validity of the pipe testing procedure also needs to be closely scrutinized.

It is convenient to classify the proposed nonenvironmental corrective actions into three groups: improved fabrication techniques, alternate materials, and residual stress improvement (RSI) techniques. The improved fabrication techniques include SHT, CRC, and HSW. Alternate materials include the low-carbon, controlled-nitrogen "Nuclear Grade" austenitic stainless steels, carbon steel, and the stabilized stainless steels. RSI techniques include IHSI and modifications of LPHSW.

There is an extensive data base on the improved fabrication techniques and the alternate materials. The data on SCC susceptibility, however, are largely based on the particular water chemistry and loading history associated with the pipe tests at GE. Although some tests on possible upset chemistry conditions are planned in the EPRI program on alternate materials, this is an area in which additional confirmatory research is needed. Similarly, additional testing on alternate loading histories is needed. Prototypic loading histories are not accurately known and would probably require prohibitively long testing times. However, the emphasis should be on lower mean values of

loading, and the effect of relatively small-amplitude cyclic loading superposed on mean loads (or slowly cycling loads) should be considered. Initial scoping studies on alternate materials can be carried out using CERT tests (varying chemistry and strain rates) and small-specimen tests (varying chemistry and loading history). Full-scale pipe testing is needed to adequately test the remedies based on improved fabrication techniques, and confirmatory pipe tests are also needed for the alternate materials. Carbon steel requires particular attention because of its widespread use in new reactor designs and in repairs of cracking-prone areas such as the core spray line. The operating experience with carbon steel piping and components has been good; however, for many of the current applications of carbon steel piping, the actual number of years of service experience is fairly limited. Although carbon steel has outperformed Type 304 stainless steel in the pipe-cracking tests, its performance in these tests is much inferior to that of other alternate materials such as the "Nuclear Grade" austenitic stainless steels.

The RSI remedies are of interest because they can be applied to existing reactors. Although these remedies are difficult to implement in an operating plant owing to high radiation levels, lack of access, etc., it is their effectiveness which is of concern here. The apparent effectiveness of residual stress based remedies (RSI and HSW) is much more strongly influenced by loading conditions than that of remedies which are based on the prevention or mitigation of sensitization. At this time, the data available from pipe tests indicate that the RSI remedies do increase the margin against IGSCC in Type 304 stainless steel. Additional data to verify this should become available from EPRI-sponsored research. EPRI also plans an analytical study to assess the possibility that the favorable residual stresses induced by the RSI will

shake down or relax in service. Since the availability of constitutive relations capable of describing the behavior of the inhomogeneous material in the HAZ under cyclic loading and long-term relaxation is highly questionable, experimental studies of the stress and strain distributions in the HAZ under cyclic loading histories are needed. (These results are also needed to benchmark the analysis used to predict the remaining lifetime of flawed piping.) The studies will also aid in understanding the relation of the stress and strain histories in the HAZs under pipe-test conditions to the deformations occurring under operating conditions.

The qualification testing planned by EPRI on the RSI remedies is focused on butt weldments in piping. However, many of the weldments encountered in practice involve different geometrical configurations. Although stress-corrosion susceptibility tests on weldments other than piping butt weldments would be difficult and extremely expensive, residual stress measurements can be carried out to benchmark the computer codes being developed in the EPRI program.

The overall program outline is shown in Fig. IV.5.

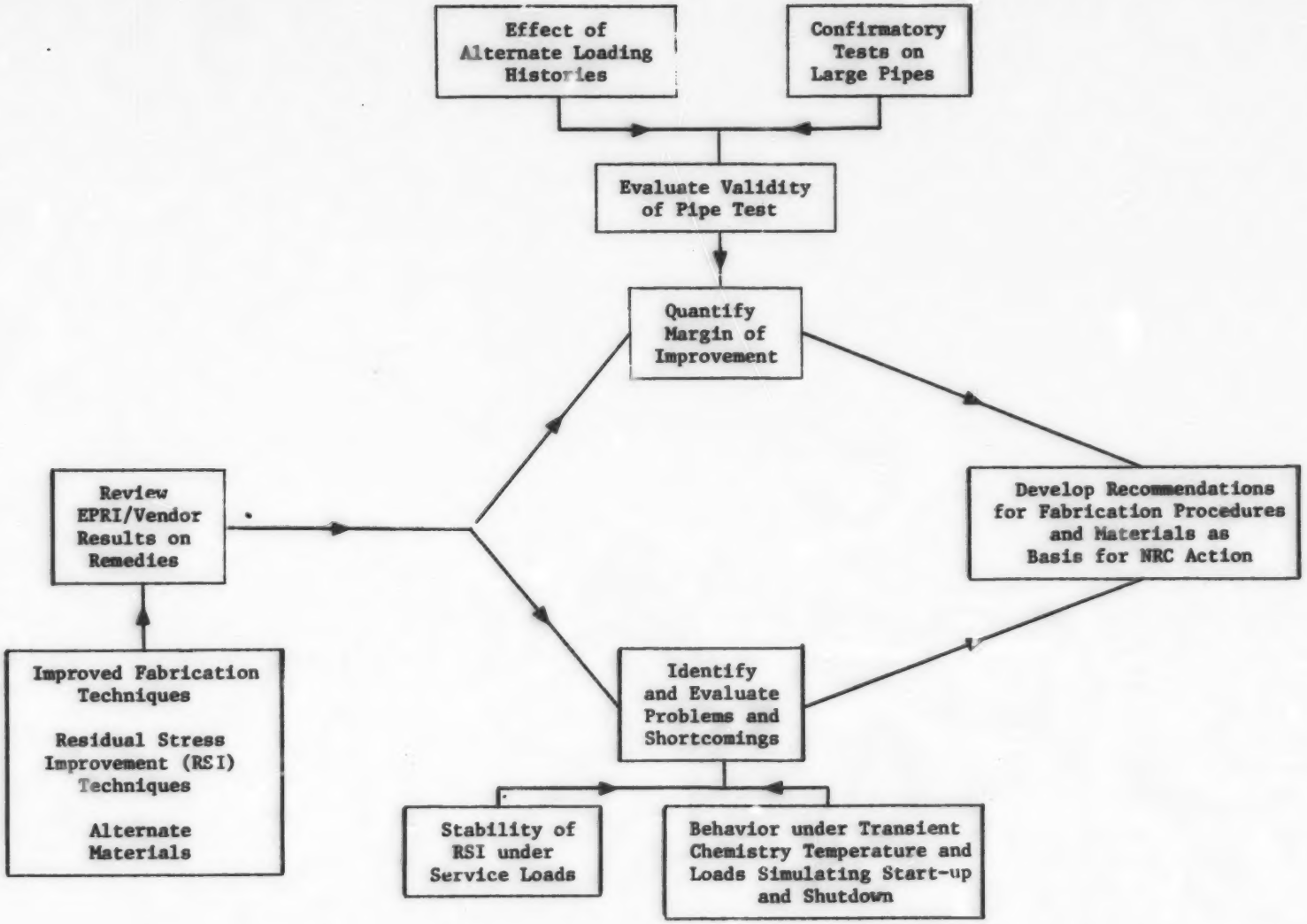

Effect of Alternate Loading Histories
Confirmatory Tests on Large Pipes
Evaluate Validity of Pipe Test
Quantify Margin of Improvement
Review EPRI/Vendor Results on Remedies
Develop Recommendations for Fabrication Procedures and Materials as Basis for NRC Action
Improved Fabrication Techniques
Residual Stress Improvement (RSI) Techniques
Alternate Materials
Identify and Evaluate Problems and Shortcomings
Stability of RSI under Service Loads
Behavior under Transient Chemistry Temperature and Loads Simulating Start-up and Shutdown

Fig. IV.5. Outline of Proposed Study of Nonenvironment Corrective Actions.

E. Environmental Corrective Actions

Deaeration during start-up and an alternate water chemistry based upon hydrogen additions to the BWR coolant appear to be within the realm of technical and economic feasibility. General Electric and Commonwealth Edison, in the second phase of a DOE-sponsored alternate water-chemistry program, have proposed a one-month test in the Dresden-2 reactor to demonstrate the hydrogen addition technique and to extend information obtained in a one-day experiment at the Oskarshamn-2 reactor, which revealed larger than expected decreases in the oxygen level of the coolant for the hydrogen addition rates employed. Although the primary motivation for development of an alternate water chemistry is mitigation of the IGSCC problem, other benefits could be derived from operation with a low-oxygen environment, e.g., a reduction in the concentration of radioactive cobalt isotopes in the coolant and the removal of crud deposits by cycling the oxygen potential of the coolant between oxidizing and reducing conditions.

Confirmatory research on the performance of materials in an alternate water-chemistry environment is required before this alternative can be implemented in operating plants. Corrosion and SCC information should be obtained from surveillance specimens during reactor demonstration tests and from more extensive laboratory experiments performed over a wide range of conditions, i.e., temperature, loading history (stress state), and water chemistry (corrosion potential as influenced by the concentrations of oxygen, hydrogen, and other species). Results from the laboratory studies can be used to (a) quantify the degree of improvement in the corrosion and IGSCC behavior of sensitized Type 304 stainless steel, low-alloy steels, and alternate

piping alloys relative to the normal BWR water chemistry, and (b) identify any detrimental effects that may arise. Since the materials should be highly resistant to IGSCC in the low-oxygen environment (particularly for the stress states associated with normal operating conditions), a mechanistically oriented research program that incorporates different experimental methods and analytical techniques will be most productive in an evaluation of the long-term performance of the materials.

Constant-extension-rate and cyclic-loading tests are needed at temperatures between 100 and 300°C on steels with various heat treatments in high-purity water in which the concentrations of various species (e.g., oxygen, hydrogen, sulfate, and chloride) are controlled and monitored. The corrosion potential of the materials in the water must also be determined. The crack nucleation and propagation characteristics of the materials should be correlated with various parameters based on the stress state, the corrosion potential, and the concentration of impurities in the water.

The experimental data will be used to develop and validate mechanistic models which then can be used to extrapolate results from accelerated laboratory tests to reactor operating conditions. These results will provide the technical basis for NRC licensing decisions relative to implementation of an alternate water chemistry in BWRs. The overall program outline is shown in Fig. IV.6.

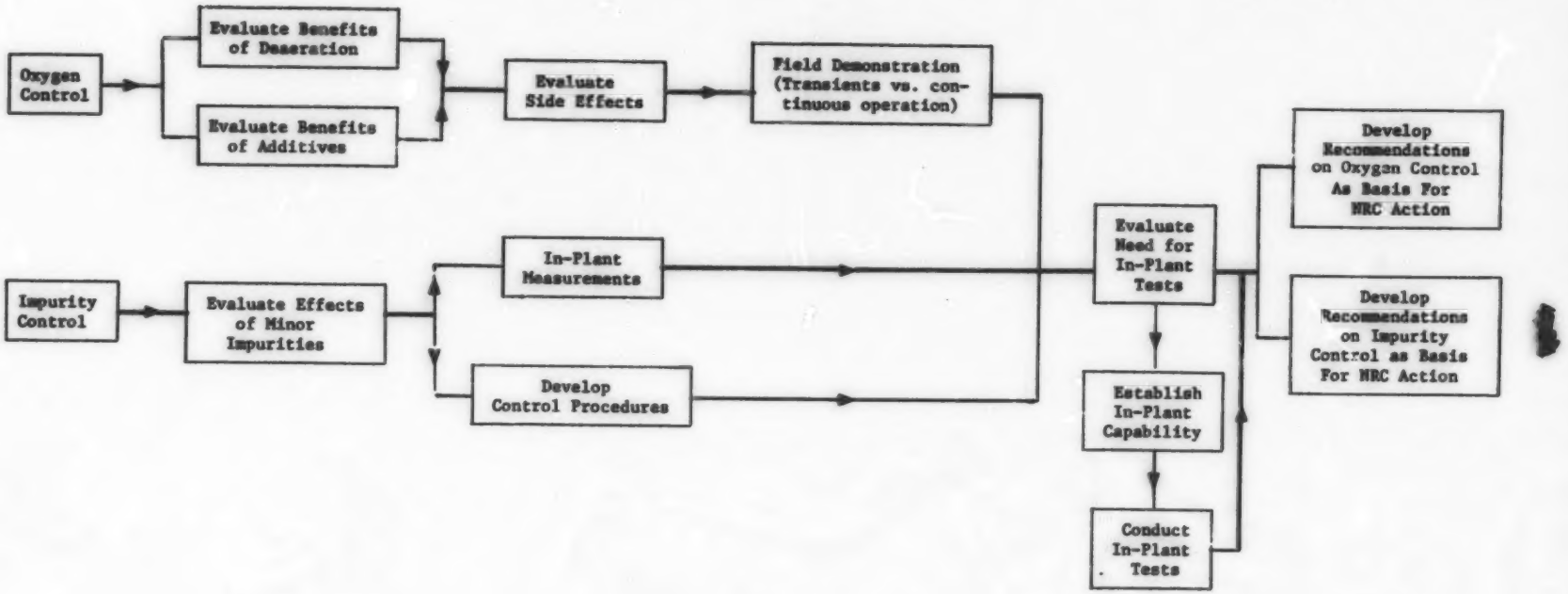

Oxygen Control
Evaluate Benefits of Deaeration
Evaluate Benefits of Additives
Evaluate Side Effects
Field Demonstration (Transients vs. continuous operation)
Impurity Control
Evaluate Effects of Minor Impurities
In-Plant Measurements
Develop Control Procedures
Evaluate Need for In-Plant Tests
Establish In-Plant Capability
Conduct In-Plant Tests
Develop Recommendations on Oxygen Control As Basis For NRC Action
Develop Recommendations on Impurity Control as Basis For NRC Action

Fig. IV.6. Outline of Proposed Study of Environmental Corrective Actions.

F. Basic Understanding of SCC

The aim of the basically oriented studies is to develop a mechanistic understanding of SCC of reactor structural steels in LWR environments. The idea is not to focus on very detailed mechanistic concepts, but to seek an overall view which provides (1) a better understanding of the potential effects (favorable and unfavorable) of proposed environmental and metallurgical corrective actions, including changes in operating procedures; (2) a more rational basis for the extrapolation of laboratory test results to predict operating plant behavior; and (3) eventually, a predictive analytical tool which can be used for the design/safety evaluation of components subjected to realistic operating histories.

The initial efforts in this highly complex area should include a survey of the available literature. This will involve assessment of the theories and models that have been proposed for general SCC phenomena, but it should concentrate on the materials of interest in LWR environments. Tentative mechanistic models can be developed and formulated analytically. Predictions of these models should be compared with available laboratory test data and refinements made. The initial phase should concentrate on the simpler static-load SCC case and consider CF only after satisfactory agreement between model and static results exists.

In order to proceed with additional development, continual interaction with experimental efforts needs to be maintained and new, mechanistically oriented experiments devised and executed. Of special interest will be the exploration of the roles of strain rate and loading mode in determining SCC susceptibility. The current programs at many other laboratories throughout the world, some of which are listed in Appendix F, should be continuously monitored.

The eventual aim is to develop a fully quantitative analytical tool which has been carefully verified against available experimental data, initially laboratory results and later data from more realistic experiments, and which can be used to predict the behavior of a component subjected to realistic load/temperature/environment histories. Though the great complexity of the phenomena involved may preclude a full realization of this goal, the ubiquitous nature of the problem, and the present rather arbitrary compartmentalizing into SCC, CF, etc., demand that a more general, mechanistically oriented approach capable of treating realistic histories be seriously attempted. The insights developed will undoubtedly aid in the devising of more meaningful experimental studies and in our understanding of materials susceptibility to SCC and the interrelation between accelerated laboratory testing and field behavior, and thus the studies should be extremely valuable whether or not a quantitatively reliable predictive model can in fact be produced.

The overall program outline is shown in Fig. IV.7.

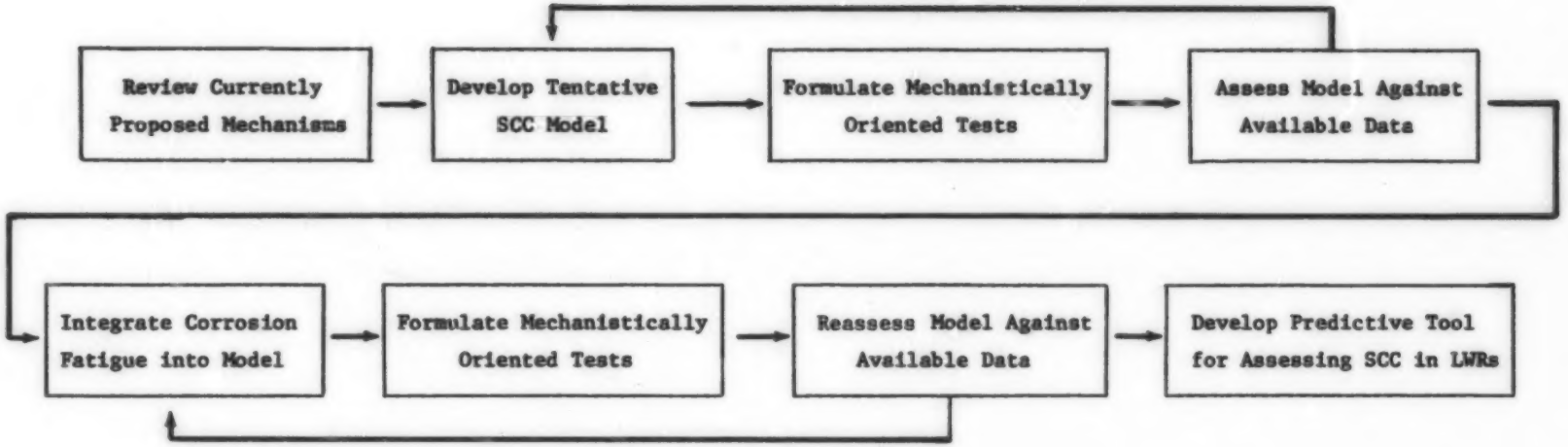

Review Currently Proposed Mechanisms
Develop Tentative SCC Model
Formulate Mechanistically Oriented Tests
Assess Model Against Available Data
Integrate Corrosion Fatigue into Model
Formulate Mechanistically Oriented Tests
Reassess Model Against Available Data
Develop Predictive Tool for Assessing SCC in LWRs

Fig. IV.7. Outline of Proposed Program of Mechanistic Studies of SCC.

APPENDIX: SUMMARY OF ONGOING PROGRAMS

A. Leak Detection and Nondestructive Evaluation

1. Field Application of Synthetic Aperture Focusing Techniques for Ultrasonic Testing of Reactor Components

Description: The objective of this program is to design and fabricate an ultrasonic imaging system based on the synthetic aperture focusing technique (SAFT). Destructive examination will be carried out on specimens to evaluate the reliability of the SAFT technique. Testing is being carried out with a completed SAFT system. Preliminary tests have shown expected improvement in flaw characterization in stainless steel specimens. A field-usable system has been constructed and evaluated using calibration blocks based on ASME Section XI codes. Tests will then be carried out on specimens (including stainless steel) that have actual flaws, including simulated service-induced flaws. Again, destructive evaluation will be used to assess the adequqacy of the technique. The specimen tests will be followed by field testing.

Organization: Southwest Research Institute

Principal Investigator: J. Jackson

Sponsor: NRC FIN#B5605

2. Improved Ultrasonic NDE of Pressure Vessels

Description: The objective is to demonstrate that a phase-sensitive synthetic array is better than a conventional pulse-echo system for characterizing a flaw or discontinuity with regard to size, shape, and orientation. An improved

*NA = Not available.

deconvolution technique has been implemented. Algorithms have been written and tested successfully for scanning pipes. Faster data processing techniques for SAFT UT have been developed leading to the possibility of "real-time" SAFT. "Spotlight scans" have been made on heavy section steel. An initial system for flaw detection has been implemented. The effectiveness of the spotlight mode and of conventional SAFT UT will be compared. The effect of various parameters on the adequacy of the SAFT UT system will be determined. New algorithms will be written to improve the system. A flaw-detection system will be developed based on methods of autoregressive time series and matched filter analysis.

Organization: University of Michigan

Principal Investigator: J. Frederick

Sponsor: NRC FIN#A4047

3. Integration of NDE Reliability and Fracture Mechanics

Description: The objectives are to establish the validity of flaw sizes in ASME Section XI standards and Evaluation Criteria, to determine the reliability and probability of flaw detection, and to develop techniques for more precise flaw location and sizing. Fracture mechanics and statistical analysis are progressing. A thermal fatigue facility is being used to grow flaws in large-diameter pipes. The digitization of ultrasonic responses representing various flaw characteristics was completed. Several chapters of a White Paper have been written. A program to establish relationships between fracture mechanics and NDE has been initiated. The level of improvement of advanced NDE techniques in comparison with conventional ones will

be quantified. The adequacy of present inspection procedures will be established by defining current state-of-the-art practice, providing samples with flaws typical of service-induced flaws, carrying out round-robin tests, and establishing the influence of material, geometry, and procedure on flaw detection, location, and sizing. The probability of detection and uncertainties in flaw characterization will be established.

Organization: Battelle-PNL

Principal Investigator: G. J. Posakony

Sponsor: NRC FIN#B2289

4. Internal-friction NDE Techniques for Detecting Incipient Cracking of Bypass Lines and Pipes in BWR Piping Systems

Description: BWR bypass lines will be monitored to relate internal friction changes to crack formation, develop an instrument package for field use, demonstrate equipment, and prepare a manual. In-house evaluation of sensor attachment has been initiated. Test samples have been evaluated under laboratory conditions and incipient cracking detected. Commonwealth Edison Dresden II feedwater lines have been instrumented. A detailed planning document is near completion. Laboratory tests for SCC monitoring are under way. The Commonwealth Edison Zion reactor will be instrumented. Baseline data for detection of IGSCC will be obtained.

Organization: Daedalean Associates

Principal Investigator: A. A. Hochrein

Sponsor: NRC

5. Measurements of Leak Rates from IGSCC

Description: The characteristics of the acoustic emissions from simulated and laboratory-grown cracks will be determined. Acoustic signatures will be sought that could form the basis for acoustic detection and location of leaks in operating nuclear plants.

Organization: Battelle-Columbus

Principal Investigator: R. P. Collier

Sponsor: EPRI-RP1554-10

6. In-service Inspection Improvement and Verification

Description: The objectives are to specify, develop, and demonstrate the adequacy of ISI for specific components including piping. This project is expected to result in the adaptation of EPRI-151 equipment developed in other programs, including ultrasonic pipe scanners and tomography systems. The sizing capability of various ultrasonic techniques will be established.

6a. Crack Tightness and Water Effects on the Ultrasonic Detection of Intergranular Stress Corrosion Cracking.

Organization: Battelle-PNL

Principal Investigator: S. Doctor

Sponsor: EPRI-T101-1

6b. Maneuverable Ultrasonic Testing Surface Scanner for Nondestructive Evaluation of Intergranular Stress-Corrosion Cracking.

Organization: Amdata Systems

Principal Investigator: D. Monahan

Sponsor: EPRI-T101-2

6c. X-ray Computed Tomography of Thick Steel Castings and Forgings.

Organization: Science Applications, Inc.

Principal Investigator: R. Polichar

Sponsor: EPRI-T101-3

6d. Development of an Ultrasonic Scanner for Inspection of Nuclear Reactor Piping and Components.

Organization: Amdata Systems

Principal Investigator: M. F. Fleming

Sponsor: EPRI-T104-4

7. UT Data-acquisition System

Description: The portable digital data-acquisition system developed in a previous effort will be replicated. The system would be available to the BWR owners group for components of opportunity. Equipment designed in RP1125 will be purchased for the BWR owners group and will be provided to experimenters and utilities for ultrasonic data acquisition. The availability of these data will help promote the use of ultrasonic NDE methods. Advanced ISI production units will become available for training purposes in 1981.

Organization: Adaptronics

Principal Investigator: A. Mucciardi

Sponsor: EPRI-T102-1

8. NDE Detection Capabilities

Description: The adequacy of improved NDE methods for ISI of austenitic piping will be established. Optimum ISI procedures to resolve potential safety issues will be determined. A large data base will be developed to verify and quantify defect-detection capabilities. Components with service-induced flaws will be used. Various NDE methods will be evaluated on a

quantitative basis, using team round-robin tests. Samples will be acquired from field failures, and flaws verified by destructive examination.

Organization: Ishikawagima-Harima Heavy Industries Co., Ltd.

Principal Investigator: NA

Sponsor: EPRI-T103

9. In-service Surveillance of Sensitive Lines

Description: The objective is to develop and demonstrate the adequacy of a surveillance system which would be capable of monitoring the growth of IGSCC in service in a quantitative way. No such system presently exists. A pipe test facility developed previously will be used to simulate BWR service conditions and initiate and propagate IGSCC for tests of the potential surveillance system. The quantitative criteria for successful monitoring of crack growth will be modified for field tests.

Organization: Battelle-Columbus

Principal Investigator: NA

Sponsor: EPRI-T104 (1980)

10. Acoustic Imaging of Stainless Steel Samples

Description: The purpose is to substantially increase the ability to find and characterize flaws in pressure vessels and piping by improved acoustic imaging techniques.

Organization: Lockheed

Principal Investigator: NA

Sponsor: EPRI-RP892-5

11. Ultrasonic System Optimization Study, Project IV, UT of Dissimilar Metal Welds

Description: The objective is to identify technique and equipment problems associated with inspection of dissimilar metal welds between reactor pressure vessel nozzle safe ends and coolant piping.

Organization: Nuclear Energy Services

Principal Investigator: L. Johnson

Sponsor: EPRI-RP892-8

12. Ultrasonic System Optimization Study, Project III, UT Systems

Description: The purpose is to aid in defining the performance of present ultrasonic inspection, in order to determine the nature of the improvements to be made. Problems in the inspection of inner surfaces of stainless steel piping and nozzles will be emphasized.

Organization: Battelle-Columbus

Principal Investigator: M. Behravesh

Sponsor: EPRI-RP892-3

13. Development of Inspection and Surveillance Methods for Primary-system Piping and Pressure Vessels

Description: The objectives are to provide pipe specimens with IGSCC, and to develop a thermal fatigue cracking technique for use in establishing the adequacy of ISI techniques for nozzles. Samples of 10- and 16-in.-dia pipe with IGSCC were produced. A test rig was developed to place thermal fatigue cracks at selected bore and bilevel radius locations in an unclad feedwater nozzle. The nozzle was successfully cracked.

Organization: General Electric Co.

Principal Investigator: J. Clark

Sponsor: EPRI-RP892-2

14. Impact Study of In-service Inspection Requirements on the Nuclear Industry

Description: The purpose is to report on the impact of in-service inspection requirements for nuclear components and structures on the nuclear industry.

Organization: General Physics Corp.

Principal Investigator: NA

Sponsor: EPRI-RP892-4

15. Application of Nonlinear Signal Processing to Pipe and Nozzle Inspection

Description: The objective is to design, fabricate, and evaluate in the field an ALN signal-processing system for inspection of austenitic pipe joints and nozzles in nuclear reactor piping systems. Under laboratory conditions, ultrasonic signals from flaws and geometrical reflectors were examined. On 4- to 26-in.-dia pipes automatic crack detection was reported to be 97% correct, with false calls accounting for 13% or less of all signals. The probability of correct detection in 10-in. or larger pipes was reported to be 100%. Close correspondence was reported between estimated and true depths for in-service cracks.

Organization: Adaptronics

Principal Investigator: A. N. Mucciardi

Sponsor: EPRI-RP1125

16. A Program to Develop a Portable Radiographic System for In-service and Repair Inspection

Description: The technology to produce a portable linear accelerator (LINAC) for use in the primary system will be developed. Many components are already available, but need to be integrated into an industrial inspection system. Testing of the integrated system was successfully completed. The performance of the LINAC will be compared with that of conventional radiography under actual field service.

Organization: Philadelphia Electric, Southwest Research, and Neil Norris

Principal Investigator: S. Wenk

Sponsor: EPRI-RP822

17. Development and Evaluation of an Acoustic Imaging System

Description: An ultrasonic system for use in real-time detection and imaging of flaws will be designed, fabricated, and field evaluated. This system will make use of scanning linear arrays of ultrasonic transducers and will meet the requirements of ASME codes. Completion is scheduled for 1979. An operating prototype was completed in 1978, primarily for heavy section steel. Ultrasonic linear array technology was demonstrated to be 50 to 100 times faster than current methods in performing ASME Section XI code inspections. Both pulse-echo and holographic image reconstruction can be used; the beam can be electronically steered.

Organization: Battelle-PNL

Principal Investigator: G. J. Posakony

Sponsor: EPRI-RP606

18. Development of a Digital Memory Acoustic Emission Zone Monitor/ Recorder for BWR Pipe Cracking

Description: The purpose is to develop an inexpensive on-line acoustic emission monitor capable of detecting the existence or growth of cracking in specific locations during plant operation, and able to discriminate against background noise. Prototype testing has indicated that the system is capable of detecting IGSCC under laboratory conditions with flow-loop and simulated reactor noise. Generally satisfactory results were obtained using GE pipe-cracking facilities and NRC cracking tests at Battelle-PNL. Tests with 26-in. piping are under way.

Organization: Battelle-PNL

Principal Investigator: P. H. Hutton

Sponsor: EPRI-RP505

19. Development of Adaptive Techniques for Pipe Inspection

Description: The objective is to develop and evaluate an ALN that will allow discrimination between geometrical and natural reflectors during an ultrasonic in-service examination.

Organization: Adaptronics

Principal Investigator: A. N. Mucciardi

Sponsor: EPRI-RP770-1

20. Development of Digital Techniques for Improved Flaw Detectability by Ultrasound Systems

Description: The possibility of reducing problems associated with ultrasonic grain-boundary scattering will be investigated. This scattering limits

the reliability of UT for coarse-grain materials such as SS weld metal. Reduction of grain echoes is expected to be achieved by varying the frequency of the transmitted signal with time and applying signal-minimization algorithms to the resulting echo signals. Computer simulation has been shown to improve flaw-to-grain echo enhancement. Improvements in flaw-to grain echo signals have been seen in titanium samples with flat-bottomed holes, using frequency agility techniques. Experimental and simulation data show that split-spectrum processing applied to a random signal correlation system may substantially enhance the visibility of flaw echoes in the presence of grain echoes. Coupling the signal-to-noise ratio enhancement advantages of the random signal flaw detection system with the flaw-to-grain echo enhancement properties of the frequency wobbling technique should greatly improve the detection of defects in materials with large grains.

Organization: Purdue University

Principal Investigator: V. L. Newhouse

Sponsor: EPRI-RP1395-4

21. Couplant Verification System

Description: The goal is to develop a couplant verification system compatible with conventional UT equipment and hand manipulation of the transducer, as well as with advanced automated scanning equipment. The system will tell the inspector whether coupling is adequate. Evaluation of couplant-monitoring signals has established a set of features that predicts the wedge couplant status. The progress to date indicates that the approach chosen will be workable in assessing couplant status. Emphasis has been placed on the

time-domain features because they are easily implemented in hardware, inexpensive, reliable in hostile environments, and compact. The remaining work on the program will be to fine-tune the terms in the preliminary algorithm used to process signals from the transducer that monitors the couplant layer.

Organization: Battelle-PNL

Principal Investigator: S. Doctor

Sponsor: EPRI-RP1448-1

22. Crack-tightness and Water Effects on the Ultrasonic Detection of IGSCC

Description: This work examines the effects of IGSCC tightness, and the effects of water within the IGSCC, on ultrasonic inspection. A test fixture was designed and fabricated to hold a standard 101.6-mm (4-in.)-dia Schedule 80 pipe. The test fixture was designed to provide loads that exceed the yield strength of the parent metal by 50%. The tight, water-filled crack studied in this program was found to have a 5-dB smaller time-domain amplitude than the air-filled crack. Thermal fatigue cracks decreased 15 dB below the no-load response when a compressive load corresponding to half the yield strength was applied. Conventional UT equipment and UT calibration must account for this fact, or they will not provide an inspection method with adequate detection sensitivity. In contrast to the time-domain amplitude information, the frequency-domain data were not altered in any significant manner by any of the variable test conditions. This indicates that advanced signal-processing methods such as the Adaptronics ALN 4000 system will not be affected by crack tightness or water. Consequently, the inspection sensitivity for detection and sizing will be constant for those variables that were employed in this study and which are typically found during in-service inspection. Field IGSCCs are typically much tighter than even the tightest

laboratory specimens. Therefore, a specimen from the field or one grown by improved laboratory techniques must be studied and compared with the results already achieved.

Organization: Battelle-PNL

Principal Investigator: S. Doctor

Sponsor: EPRI-RP1448-1

23. Development of Prototype Advanced Equipment for Manual UT Investigations

Description: The objective is to define and develop advanced prototype instrumentation or techniques which are applicable to UT and surveillance of reactor pipe cracking. A current area of emphasis is the development of an IGSCC "call-confirmer", an auxiliary aid to manual ultrasonic pipe inspection conducted according to ASME XI code. The reliability of IGSCC detection depends on the ability to discriminate geometric reflectors from IGSCC. Two approaches to discrimination are geometric, based on search-unit design, and a combination of automation and frequency-spectrum analysis using digital processing of ultrasonic signals. In this project, features of both approaches which were judged to be consistent with conventional, code-acceptable, manual UT approaches were selected. The scope of this project was to develop, implement, and perform preliminary evaluations of a prototype "call-confirmer". These tasks were completed in 1979 for one of the prototypes evaluated in the Engineering Evaluation Test (see RP892-3). Physical and functional modification of the unit will depend on the confirming results obtained.

Organization: Zeger-Abrams, Inc.

Principal Investigator: A. Zeger

Sponsor: EPRI-RP1448-5

24. Quick-reaction Capability for Manufacture of Flawed Specimens

Description: The objective of this project is to provide a capability to observe and monitor the growth of IGSCC in large pipes under conditions that replicate BWR service. A secondary objective is to rapidly produce IGSCC pipe specimens in support of alternate projects. Artificial crevice-inducing material is used to initiate IGSCC. The initial test rig has been completed and is in a testing stage. Baseline NDE information (UT, radiography) has been obtained; monitoring instrumentation includes strain gauges and acoustic emission monitors. However, several welds believed to represent the extremes of residual stress obtained with conventional welding procedures are available for testing. Future weld specimens will also be made in duplicate; the residual stress levels and distribution of one weld specimen will be established by destructive assay.

Organization: Battelle-PNL

Principal Investigator: R. Clark

Sponsor: EPRI-RP1448-6

25. Production of Controlled IGSCC in Large-diameter Pipe

Description: The feasibility of producing IGSCC with controlled size and location in large pipes will be demonstrated. These specimens may serve as representative samples and as calibration standards, and may eliminate the need for destructive assay to confirm the sizing capability of proposed ISI methods. In this technique, the crack-initiation period

is shortened by the introduction of an applied artificial crevice-inducing material. IGSCC growth is achieved without applied mechanical stress in a high-purity oxygenated-water environment similar to that encountered in BWR service; the resulting cracks are "tight" and are believed to be replications of those obtained under service conditions. To date, ten 12-in. Schedule 100 samples have been produced. All cracks are "tight", location control is essentially perfect, and crack depths of 5-20% of wall thickness have been achieved. The scatter in depths is substantial. The specimens are currently in use in verification programs. The prospects for better depth control are being investigated.

Description: Ishikawagima-Harima Heavy Industries Co., Ltd.

Principal Investigator: T. Kawamoto

Sponsor: EPRI-RP1448-7

26. Nondestructive Evaluation

Description: Ultrasonic techniques will be used to quantitatively describe the size, shape, and orientation of flaws in anisotropic materials (includes stainless steel weld metal). The main factors affecting the apparent attenuation of waves propagating in anisotropic materials with orthotropic symmetry have been identified. The beam-deviation phenomenon has been measured experimentally for longitudinal and shear waves. Agreement between calculation and experiment was good. Consequences for flaw detection in stainless steel weld metal have been established. Studies on bicrystals have been initiated to further understand the interactions of longitudinal and shear waves at grain boundaries.

Organization: Argonne National Laboratory

Principal Investigator: K. J. Reimann

Sponsor: DOE

27. Nondestructive Evaluation

Description: The goal is to develop the boundary integral equation (BIE) method for predicting the ultrasonic scattering of two-dimensional flaws in solids. Good agreement has been found between the known solution for a cylindrical void and first-generation results from the BIE methods. The use of dynamic photoelasticity for the characterization of ultrasonic waves interacting with surface-breaking waves has been initiated.

Organization: Ames Laboratory

Principal Investigator: C. P. Burger

Sponsor: DOE

28. Elastic Wave Scattering and Quantitative Flaw Identification

Description: The objective is the development of an analytical, scientific reference data base of scattering calculations for flaw identification. The integral equation method will be used. Scattering will be calculated for special geometries by various approximations, and compared with exact results using a sphere.

Organization: Los Alamos Scientific Laboratory

Principal Investigator: J. E. Gubernatis

Sponsor: DOE

29. Acoustic Emission Signature Analysis

Description: Acoustic emission techniques will be applied to detect cracking mechanisms in metals and crack growth in embrittled steels. Multiple-transducer Fourier frequency analysis of acoustic emissions, and the fracture momentum concept, will also be investigated.

Organization: Rockwell International Science Center

Principal Investigator: O. Buck

Sponsor: DOE

B. Analysis of Sensitization

A number of ongoing programs to study sensitization in austenitic stainless steel and alternative alloys have been sponsored by EPRI, NRC, and DOE. The scope of the studies includes a basic understanding of sensitization, detection methods, influence of fabrication variables, and correlation with IGSCC susceptibility. The data base for quantitative correlation of DOS and IGSCC susceptibility has been obtained primarily from CERT tests, although some full-scale pipe test data are also available. IGSCC propagation rates measured from fracture-mechanics-type specimens may soon be available for the correlation.

1. Basic Studies on the Variability of Fabrication-related Sensitization Phenomena in Stainless Steel

Description: The susceptibility of Type 304 stainless steel to IGSCC will be related to the microstructural changes accompanying both normal sensitization and LTS. Welds from pipe, steel subjected to simulated weld thermal/strain transients, and a synthesized alloy having the near-grain-boundary chemistry of sensitized Type 304 stainless steel will be examined and compared to determine how closely their corrosion behavior follows the above microstructural relationship. Determinations of the corrosion rates, anodic polarization behavior, and crevice environment in a simulated crack will complement the above work, and provide a basis for alternate corrosion test environments. Finally, the detailed understanding developed in this work will be extended in scoping studies on a variety of other stainless steels, such as 304LN, 316 and 347.

Organization: General Electric Co.

Principal Investigator: P. L. Andresen et al.

Sponsor: EPRI (RP1072-1)

2. Low-temperature Sensitization - Phase 1

Description: The effects of LTS on crack initiation and propagation are to be quantified. The crack propagation rate will be determined as a function of environment (oxygen content and temperature) and LTS heat treatment. A model for LTS and an experimental method for identifying an LTS-susceptible heat will be developed.

Organization: Stanford Research Institute

Principal Investigators: D. D. Cubicciotti and S. S. Wing

Sponsor: EPRI (RP1565-1)

3. Low-temperature Solute Segregation

Description: The influence of impurities on LTS of Type 304 SS will be examined. The time-temperature-impurity dependence of Type 304 SS piping in the as-welded + LTS condition will be determined. Effects of the segregated species on the initiation and propagation of IGSCC will be determined by collaboration with Program 1565-1.

Organization: Rockwell International Science Center

Principal Investigator: J. Lumsden

Sponsor: EPRI (RP1565-2)

4. Localized Corrosion and SCC Behavior of Austenitic Stainless Steel Weldments Containing Retained Ferrite

Description: The corrosion behavior of stainless steels containing welds, and stainless steel weldments, will be studied with particular

attention to pitting and SCC in chloride-containing solutions. Pitting corrosion will be studied at room temperature and 290°C in pressure vessels, utilizing potentiodynamic and galvano-kinetic test procedures coupled with optical and electron metallography. Other areas of investigation include the effect of P and S on pitting corrosion; scanning Auger electron spectroscopy on austenite/ferrite interfaces; slow-extension-rate SCC experiments; and analytical electron microscopy.

Organization: Rensselaer Polytechnic Institute

Principal Investigator: W. F. Savage, D. J. Duquette

Sponsor: DOE/DBES (01-1)

5. Intergranular Stress Corrosion

Description: Studies will be performed on the electrochemistry of surfaces of iron and nickel-base alloys at high temperatures, using an ac polarization technique. A scanning ac polarization technique will be developed. Corrosion currents from a propagating stress-corrosion crack will be measured using a scanning reference electrode technique. Chromium depletion and grain-boundary segregation in stainless steels and Inconel 600 will be determined using energy dispersive x-ray analysis in conjunction with scanning-transmission electron microscopy. The effect of temperature on SCC of sensitized stainless steel in oxygenated high-temperature water will be assessed.

Organization: Brookhaven National Laboratory

Principal Investigators: J. R. Weeks, Brijesh Vyas, M. W. Kendig. H. S. Isaacs, C. S. Pande

Sponsor: DOE/DBES (01-2)

6. High-temperature Chemistry and Thermodynamics of Structural Materials

Description: The thermodynamics and kinetics of formation and dissolution of carbide and nitride microphase precipitates will be investigated.

Organization: Oak Ridge National Laboratory

Principal Investigators: J. T. Bell et al.

Sponsor: DOE/DBES (03-2)

7. Primary Coolant Rupture Study

Description: This study is concerned in part with the development of EPR test techniques that can be used in both service and construction phases for evaluating the susceptibility to IGSCC of BWR stainless steel components. The EPR data base will be broadened so that the EPR techniques and evaluation can be applied to welded components with a high level of confidence.

Organization: General Electric Co.

Principal Investigator: W. L. Clarke

Sponsor: NRC (FIN #B3012)

C. Analysis of Crack Growth

1. Crack Initiation

(a) Engineering Models for Predicting SCC of BWR Piping Welds

Description: A literature review of field and laboratory data on IGSCC of sensitized Type 304 stainless steel will be written. Mechanistic and statistical descriptions of the nature of the crack initiation processes will be reviewed. Further areas of needed research will be identified.

Organization: Science Applications, Inc.

Principal Investigator: D. O. Harris

Sponsor: EPRI (RP1554-6)

(b) Assessment of Some System Consequences of IGSCC in BWR Piping

Description: The objectives are to assess the system consequences of coolant leakage due to IGSCC, including the probability of multiple coolant leaks; to develop an improved stress rule index; and to review the data base of IGSCC field experience.

Organization: S. Levy, Inc.

Principal Investigator: T. L. Gerber

Sponsor: EPRI (T118-12)

2. Crack Growth

(a) Crack Propagation Rate Studies on Type 304 Stainless Steel in a BWR Environment

Description: IGSCC crack growth rates will be measured in BWR environments using standard LEFM 1T compact tension test specimens.

Organization: Argonne National Laboratory

Principal Investigator: J. Y. Park

Sponsor: EPRI (T-117-1, formerly RP449-1)

(b) Analysis to Predict the Shapes of Growing SCCs in BWR Piping Welds

Description: This investigation focuses on analytical studies to determine the major parameters that influence crack shape in Type 304 stainless steel undergoing IGSCC, and on corrective actions necessary to ensure that cracks have small aspect ratios and exhibit leak-before-break behavior.

Organization: Aptech Engineering

Principal Investigator: W. P. McNaughton

Sponsor: EPRI (RP1554-7)

(c) The Growth and Stability of SCCs in Large-diameter BWR Piping

Description: IGSCC growth rates in BWR environments will be determined for base metal and weldments. Residual stress distributions will be measured using boiling $MgCl_2$. Late in 1980, full-scale pipe experiments will be initiated to verify crack-growth models. Failure analysis diagrams will be developed for flawed piping based on a critical net section stress criterion. Late in 1980, work will be begun on the development of a dynamic analysis of flawed piping. Dynamic pipe tests will be carried out (post-1980).

Organization: General Electric Co.

Principal Investigator: R. Horn, S. Rangsnath

Sponsor: EPRI (RP1554-1)

(d) Corrosion-fatigue Characterization of Reactor Pressure Vessel Steels

Description: Corrosion-fatigue crack growth rates for both irradiated and unirradiated material in PWR environments will be measured experimentally. An analytical model based on a mechanistic description of the corrosion processes at the crack tip will be developed to better understand the interaction of corrosion and fatigue crack growth.

Organization: Babcock and Wilcox, Westinghouse, others (not yet identified)

Principal Investigator: NA

Sponsor: EPRI (RP1325-1 B&W, RP1325-4, W)

(e) BWR Environmental Cracking Margins for Carbon Steel Piping

Description: The objectives are to examine the behavior of carbon steel in BWR environments and to develop design rules to quantify environmental effects on crack initiation and growth. Carbon steel piping and weldments will be tested in simulated BWR environments. Crack growth rates will be measured, and K_{ISCC} determined. Low-cycle fatigue crack initiation in BWR environment will be studied. Design stress rules will be formulated to quantify environmental effects on crack initiation and growth.

Organization: General Electric Co.

Principal Investigator: D. Weinstein

Sponsor: EPRI (RP1248-1)

3. Test Procedures for J-R Curves and Determination of J-R Curves for Materials of Interest in Nuclear Primary Coolant Systems

(a) Rapid J-R Curve Evaluation

Description: "Key curves" will be used to evaluate J-R curves directly from the load-displacement record without direct independent measurement of crack length. Sub-size specimens will be tested to develop "key curves" for larger specimens. 1T-CT specimens of A533B will be tested to develop a J-R curve for rapid loading of A533B steel at 300°C.

Organization: U.S. Naval Academy

Principal Investigator: J. A. Joyce

Sponsor: NRC

(b) J-R Curve Testing of Reactor Pressure Vessel and Primary Piping Steels

Description: A J-R curve data base will be developed for reactor and piping steels and piping weldments. Tearing instability predictions for compact specimens will be verified. Limits of J-controlled crack growth will be explored experimentally. Unloading compliance techniques will be used to characterize the J-R curves at RT, 300°F, and 550°F for A106-A516 and A351-CF-8A piping alloy weldments. Tearing instability predictions for compact specimens will be tested by varying the testing machine compliance. Compact specimens with a range of thicknesses and ligament widths will be tested to determine limits for J-controlled growth.

Organization: Naval Ship Research and Development Center

Principal Investigator: J. P. Gudas

Sponsor: NRC

(c) Structural Integrity of Water Reactor Pressure Boundary Components

Description: Objectives include the development of fracture toughness criteria including effects of irradiation; cyclic crack-growth rate studies in a reactor environment; and characterization of metallurgical factors affecting irradiation sensitivity, postirradiation properties recovery, and sensitivity to reirradiation. J-R curves for irradiated pressure vessel materials will be measured. Cyclic crack growth data will be obtained on unirradiated and irradiated pressure vessel materials and piping steels and associated weldments.

Organization: Naval Research Laboratory

Principal Investigator: F. J. Loss

Sponsor: NRC

4. Experimental Verification of Current Elastic-Plastic Fracture Methodology

(a) Heavy Section Steel Technology Program

Description: (FY 1980) The objectives are a realistic evaluation of fracture potential and development of fracture-prevention criteria for heavy section reactor vessels. Finite elements available for evaluation of stress intensity factors in complex geometries will be studied. A simplified elastic-plastic fracture methodology will be developed. Crack arrest in reactor vessels and irradiation effects on toughness will be studied. Thermal shock tests will be performed, and a vessel containing low-ductile shelf material for validation of elastic-plastic fracture prediction will be constructed.

Organization: Oak Ridge National Laboratory

Principal Investigator: G. D. Whitman

Sponsor: NRC

(b) Degraded Piping

Description: Tests will be carried out to quantify the capability of cracked piping to resist fault-loading conditions. Experimental results will be compared with tearing-instability analyses. Preliminary tests will be conducted on cracked piping with artificial or field-induced flaws under high bending or axial loads, and results compared with tearing-instability predictions.

Organization: U.S. Naval Ship Research and Development Center

Principal Investigator: J. Gudas

Sponsor: NRC

(c) Instability Predictions for Circumferentially Cracked Large-diameter Type 304 Stainless Steel Pipes under Dynamic Loads

Description: Development and experimental validation of a pipe fracture analysis model will be carried out. Elastic-plastic fracture methodology will be used to develop a piping fracture analysis model. Quasi-static (1980) and dynamic (1980-81) pipe fracture experiments will be carried out.

Organization: Battelle-Columbus

Principal Investigator: M. Kanninen

Sponsor: EPRI (RP1554-2)

(d) Application of Tearing Modulus Stability Concepts to Nuclear Piping

Description: Piping tests will be conducted to verify the validity of the tearing modulus stability criterion. Tests will also be conducted to verify that a tearing modulus developed from compact-tension specimens is applicable to real structures. Currently available tearing modulus analyses of piping will be reexamined to determine the range of applicability.

Organization: Washington University

Principal Investigator: P. Paris

Sponsor: EPRI

(e) Effects of Specimen Size and Configuration on Fracture Toughness and Ductile Instability

Description: The goal is to provide the technology for a quantitative assessment of the structural integrity of cracked structures of ductile materials. Activities include the experimental assessment of the effect of specimen dimensions on J-R curves; assessment of limits on J-controlled crack growth; and experimental verification of elastic-plastic fracture methodology on specimens of differing geometrical configuration and size.

Organization: Westinghouse

Principal Investigator: J. Landes

Sponsor: EPRI (RP1238)

5. Application of Elastic-Plastic Fracture Methodology to Structural Systems and Further Analytical Development

(a) Flawed Stainless Steel Piping Stability Analysis

Description: J- and COD-resistance curve techniques will be applied to the prediction of stability of flawed piping. Piping systems will be analyzed to determine "effective compliance". Verification tests on 4- and 10-in. piping will be conducted.

Organization: General Electric Co.

Principal Investigator: C. F. Shih

Sponsor: EPRI (RP1554-8)

(b) A Combined Macroscopic and Microscopic Approach to the Fracture of Metals

Description: Macroscopic fracture mechanics will be correlated with microstructurally associated deformation and failure mechanisms in metals. Numerical analysis of stable crack growth under small-scale yielding will be performed, and correlations and comparisons made with tearing modulus values. Void formation and linkage at grain boundaries and enhancement by hydrogen environments will be studied.

Organization: Brown University

Principal Investigator: J. R. Rice

Sponsor: DOE (Div. of Physical Research)

(c) Advanced Fracture Mechanics Methodology

Description: The goal is to develop methodology required to perform safety analyses under conditions in which stable crack growth occurs. J-R curves will be determined by load displacement analysis. Tearing instability analyses will be verified for specimens of different sizes and geometrical configurations. Limits of J-controlled crack growth will be assessed.

Organization: Washington University

Principal Investigator: P. C. Paris

Sponsor: NRC

(d) Crack Arrest Technology

Description: The objective is a characterization of crack arrest as applied to primary systems of LWRs. Fracture toughness of pressure-vessel material will be measured. HSST thermal shock experiments will be analyzed, and elastic-plastic fracture methodology will be surveyed and reviewed.

Organization: Battelle-Columbus

Principal Investigator; A. Rosenfield

Sponsor: NRC

D. Nonenvironmental Corrective Actions

1. Induction Heating Stress Improvement (IHSI)

Description: Experimental studies will be performed to optimize and verify induction-heating parameters and water-cooling requirements and determine residual stress distributions for IHSI of 4-, 10-, 16-, and 26-in. diameter piping. Environmental pipe tests will be carried out to quantify the degree of IGSCC immunity produced by IHSI treatments. Effects of IHSI on precracked pipes will be determined.

Organization: General Electric Co.

Principal Investigator: D. Bertossa

Sponsor: EPRI (RP1394-1)

2. Computational Analysis and Experimental Evaluation of Residual Stresses from Induction Heating

Description: The objectives are development of an analytical model for the IHSI process; verification of the model by comparing predicted temperatures and residual stresses with values obtained by experiment; modeling of the stress predistribution during the pipe testing and during service shakedown under cycles of typical operating conditions; analytical study of the effects of IHSI on crack growth behavior; and experimental measurement of residual stress distributions in 10- and 26-in. Schedule 80 pipe weldments.

Organization: University of Tulsa

Principal Investigator: E. F. Rybicki

Sponsor: EPRI (RP1394-4)

3. IHSI Report Preparation

Description: A technical report will be prepared describing the research and development work on IHSI carried out by (and sponsored by) Ishikawagima Harima Heavy Industries Co. (IHHI) in Japan.

Organization: Ishikawagima Harima Heavy Industries Co.

Principal Investigator: NA

Sponsor: EPRI (RP1394-5)

4. Heat Sink Welding (HSW)

Description: An analytical model will be developed to predict the effect of welding parameters, such as the heat input, number of passes, and pipe diameter, on the residual stress distribution associated with HSW. The model will be used to determine optimal parameters for HSW. Heat sink welded pipes will be prepared; experimental measurements of temperature and residual stress will be obtained and compared with the predictions of the model.

Organization: Battelle-Columbus

Principal Investigator: M. F. Kanninen

Sponsor: EPRI (RP1576-1)

5. Qualification of Last Pass Heat Sink Welding (LPHSW) Using Stagnant Water Cooling

Description: The residual stress distribution in large-diameter piping produced by LPHSW with stagnant water cooling on the inner surface

will be determined. A 12- and 28-in. combination pipe cutter, in which separation and contouring are accomplished in a single mode, will be evaluated. Experimental welds will be fabricated using LPHSW with stagnant water cooling. Both manual and automatic welds will be made. Experimental measurements of the residual stresses associated with each weldment will be made.

Organization: Boston Edison

Principal Investigator: R. DeMuth

Sponsor: EPRI (RP1576-2)

6. Parametric Studies for Stress Corrosion in Type 304 Stainless Steel

Description: The effects of changes in the applied stress, oxygen level, and cyclic frequency on SCC in the EPRI/GE pipe test will be determined.

Organization: General Electric Co.

Principal Investigator: T. P. Diaz

Sponsor: EPRI (T112-1)

7. Qualification and Implementation of Alternate Piping Alloys for BWR Application

Description: Qualification of piping materials which are resistant to sensitization and immune to IGSCC under BWR service conditions will be carried out. This program, started in 1977, includes the selection of candidate material, initial screening tests, and the choice of a prime and backup candidate material. Pipe tests were used for the initial screening, and statistical pipe tests are planned to qualify the final alloy choice.

Organization: General Electric Co.

Principal Investigator: J. Alexander

Sponsor: EPRI (RP968-1)

8. BWR Environmental Cracking Margins for Carbon Steel Piping

Description: The objectives include testing of carbon steel piping and weldments in simulated BWR environments; crack growth rate measurements and determination of K_{ISCC}; a study of low-cycle fatigue crack initiation in BWR environments; and formulation of design stress rules to quantify environmental effects on crack initiation and growth.

Organization: General Electric Co.

Principal Investigator: D. Weinstein

Sponsor: EPRI (RP1248-1)

E. Environmental Corrective Actions

The General Electric Company, in conjunction with the Commonwealth Edison Company, is investigating the technical feasibility and cost associated with alternate water chemistries for mitigating the incidence of SCC of austenitic stainless steels in BWRs. The effects of volatile feedwater additives on plant operations and the performance of components and materials are being evaluated through measurements at the Dresden-2 and Nine Mile Point-1 Nuclear Power Stations and in laboratory facilities. This program is funded by the U.S. Department of Energy. The Electric Power Research Institute also sponsors programs on the measurement and control of oxygen in the reactor coolant during startup, the effect of the environment on the SCC of sensitized stainless steels under simulated conditions, and mechanical-environmental interactions in carbon steels under simulated BWR water-chemistry conditions. Similar work is in progress in Japan and other countries. The information developed in these programs is relevant to the development of an alternate water chemistry for operating conditions in BWRs.

1. Oxygen Suppression in Boiling Water Reactors

Description: The purpose is an in-depth engineering evaluation of oxygen suppression additives supported by critical experiments, where required, to resolve substantive uncertainties. The optimum oxygen-suppression approach for a 3-year demonstration experiment in a BWR plant will be selected.

Organization: General Electric Co.

Principal Investigator: E. L. Burley

Sponsor: DOE/NPD

2. BWR Water Chemistry Monitoring

Description: The objectives are to design, fabricate, and install the necessary facilities to monitor the electrochemical potential of stainless steel without affecting reactor operations.

Organization: Nuclear Water and Waste Technology Corp.

Principal Investigator: W. Pearl

Sponsor: EPRI (RP706-1-3)

3. Evaluation of BWR Resin Intrusions on SCC of Reactor Structural Materials

Description: SCC tests will be performed on Type 304 stainless steel in the as-received and welded condition with a low-temperature sensitization treatment to isolate the effect of chemical intrusions on initiation and propagation of SCC.

Organization: Battelle-PNL

Principal Investigator: B. Francis

Sponsor: EPRI (RP1563-1)

4. Effect of Deaeration during Startup of BWR on SCC of Type 304 Stainless Steel

Description: The objective is to determine whether deaeration during startup of BWRs will be beneficial in reducing SCC of Type 304 stainless steel. Water from a Swedish BWR will be used to test as-welded and as-welded plus sensitized Type 304 stainless steels which do and do not meet ASEA-ATOM specification with respect to a standard intergranular corrosion test.

Organization: ASEA-ATOM (Sweden)

Principal Investigator: P. Fejes

Sponsor: EPRI (RP701-4)

5. Effects of BWR Deaeration on the Corrosion of Carbon and Low-alloy Steels

Description: The purpose is to investigate the combined effect of oxygen content and temperature of high-purity BWR-type water on the corrosion and SCC behavior of plain carbon and low-alloy steels under steady-state (operating) conditions and transient conditions associated with start-up and shutdown of BWRs.

Organization: Ohio State University

Principal Investigator: NA

Sponsor: EPRI (RP701-5)

6. Investigation of the Benefits of BWR Oxygen Control

Description: The objective is to generate data which may be used to convince utility management that start-up deaeration will reduce SCC of Type 304 stainless steel piping. The resistance of the sensitized HAZ of welds to SCC will be measured by CERT, constant-load, and cyclic-load tests with temperature ramping in simulated reactor start-up environments.

Organization: General Electric Co.

Principal Investigator: NA

Sponsor: EPRI (RP701-3)

7. BWR Environmental Cracking Margins for Carbon-Steel Piping

Description: Carbon steel cracking margins in simulated BWR environments will be defined by means of tests on large pipes and experimental work, to provide data for the development of design stress rules for a near-term crack initiation criterion and a longer-range crack growth/fracture mechanics analysis.

Organization: General Electric Co.

Principal Investigator: D. Weinstein

Sponsor: EPRI (RP1246-1)

F. Basic Understanding of SCC

Very few SCC programs are basically or mechanistically oriented. However, some of the more applied programs have certain elements of key interest to our study. Naturally, we shall attempt to maintain some familiarity with all SCC studies worldwide, but we list here the principal domestic programs that will be followed most closely.

1. Localized Corrosion and SCC Behavior of Austenitic Stainless Steel Weldments Containing Retained Ferrite

Description: Corrosion behavior of stainless steels containing welds and stainless steel weldments will be examined with particular attention to pitting and SCC in chloride-containing solutions. Pitting corrosion will be studied at room temperature and 290° in pressure vessels utilizing potentiodynamic and galvanokinetic test procedures coupled with optical and electron metallography. Investigations will encompass the effect of P and S on pitting corrosion; scanning Auger electron spectroscopy on austenite/ferrite interfaces; slow extension rate stress corrosion cracking experiments; and analytical electron microscopy.

Organization: Rensselaer Polytechnic Institute

Prinipcal Investigator: W. F. Savage, D. J. Duquette

Sponsor: DOE/BES

2. Fundamental Studies of Stress Corrosion and Corrosion Fatigue Mechanisms

Description: The mechanisms controlling SCC and corrosion fatigue cracking of iron, iron-chromium-nickel, and nickel-based alloys in gaseous and aqueous environments will be investigated. Computer modeling and experimental measurement of surface and grain-boundary segregation of S,

P, Sb, C and O in Fe and Ni will be performed. Phenomena such as fracture mode changes and crack growth rates will be studied as a function of grain-boundary chemistry and electrochemical potential. Effect of plastic strain and various gaseous environments on the quantity and distribution of surface adsorption will be studied in an Auger electron spectrometer using an in-situ straining stage.

Organization: Battelle-PNL

Principal Investigators: R. H. Jones, M. T. Thomas, S. M. Bruemmer

Sponsor: DOE/BES

3. Mechanisms of SCC

Description: Intergranular and transgranular crack propagation in engineering materials will be studied using fractographic (SEM, TEM), metallographic, and acoustic-emission measurements. The role of hydrogen in the cracking process will be examined.

Organization: University of Illinois

Principal Investigator: E. N. Pugh

Sponsor: DOE/BES

4. Corrosion Processes and Energy Systems

Description: A multidisciplinary corrosion research center will be created to study fluid flow effect; inhibitors; water-chemistry effects; behavior of ceramic materials; corrosion in liquid metal systems; surface interactions with mechanical properties; protective films on metals;

atmospheric corrosion of metals, polymers and glasses; and concentrated and complex electrolytes. Technology transfer will be accomplished through the preparation of movies, handbooks, and assessments in specific areas of corrosion.

Organization: University of Minnesota

Principal Investigator: R. W. Staehle

Sponsor: DOE/BES

5. Localized Corrosion and Stress-cracking Phenomena Related to Energy Technologies

Description: This program will focus on basic electrochemical investigations of corrosion reaction mechanisms operative in localized attack on metals (e.g., titanium, stainless steel). The goal is an increased understanding of corrosion in active and passive states; effects of restrictive geometries (pitting, crevice corrosion, SCC); kinetics of coupled active-passive electrode systems; kinetics of corrosion reactions in concentrated aqueous electrolytes; and effect of strain on dissolution kinetics. Rapid electrochemical methods for testing susceptibility to localized attack will be developed.

Organization: Oak Ridge National Laboratory

Principal Investigators: F. A. Posey, A. L. Bacarella, E. J. Kelley, A. A. Palko

Sponsor: DOE/BES

6. A Study of Grain-boundary Segregation Using the Auger Electron Emission Technique

Description: The study will encompass grain-boundary segregation in metals and the effect on properties; theoretical and experimental aspects

of SCC; Auger electron spectroscopy; segregation of S and Bi in Fe and S in Mo; SCC of aluminum bronzes; hydrogen embrittlement of copper alloys and pure iron; liquid and solid metal embrittlement as affected by grain-boundary segregation; and embrittlement of P-bearing Ni-Cu alloys by liquid Hg.

Organization: Michigan Technological University

Principal Investigators: D. F. Stein, L. A. Heldt

Sponsor: DOE/BES

7. Intergranular Stress Corrosion

Description: The electrochemistry of surfaces of iron- and nickel-base alloys at high temperatures will be studied using an ac polarization technique. A scanning ac polarization technique will be developed. Corrosion currents from a propagating stress corrosion crack will be measured using a scanning reference electrode technique. Chromium depletion and grain-boundary segregation in stainless steels and Inconel 600 will be determined using energy dispersive x-ray analysis attached to a scanning transmission electron microscope. The effect of temperature on the SCC of sensitized stainless steel in oxygenated high temperature water will be assessed.

Organization: Brookhaven National Laboratory

Principal Investigators: J. R. Weeks, Brijesh Vyas, M. W. Kendig. H. S. Isaacs, C. S. Pande

Sponsor: DOE/BES

8. Temper Embrittlement and Stress Corrosion Studies

Description: The temper-embrittlement phenomenon observed in alloy steels is being studied using Charpy tests and Auger-SIMS scans on a series of 3140 steel samples. Results indicate a difference in the distribution of sulfur in the embrittled and deembrittled samples, which may be of significance to the understanding of the fundamental cause of the phenomenon. Stress-corrosion studies are being initiated; microstrain experiments have been started on single crystals of high-purity alpha-brass.

Organization: Ames Laboratory

Principal Investigators: C. V. Owen, S. Shei, T. E. Scott

Sponsor: DOE/BES

9. Mechanisms of Environmental Cracking in Power Generation Systems

Description: In-service cracking of power-system components (such as steam turbines, pipes, and nozzles) results in costly unscheduled outages for repair and correction. The objective is to advance knowledge of potential cracking phenomena in ferrous alloys to the stage at which cracking rates can be predicted from measurable parameters of fundamental importance to the cracking mechanisms.

Organization: General Electric Co.

Principal Investigator: F. P. Ford and M. Silverman

Sponsor: EPRI RP1332-1

10. Fracture of Nuclear Reactor Piping

Description: The University of Washington will perform a detailed analytical study of subcritical flaw growth in some of the critical nuclear piping regions and a theoretical investigation of the arrest of propagating brittle cracks. A preliminary analytical investigation of the depressurization phenomena in a high-pressure, high-temperature water system to supplement the crack growth and arrest problem will be included. This is a 2-year effort.

Organization: University of Washington

Principal Investigator: NA

Sponsor: EPRI

11. Basic Studies on the Variability of Fabrication-related Sensitization Phenomena in Stainless Steels

Description: The susceptibility of Type 304 stainless steel to IGSCC will be related to the microstructural changes accompanying both normal sensitization and low-temperature LTS. Welds from pipe, steel subjected to simulated weld thermal/strain transients, and a synthesized alloy having the near-grain-boundary chemistry of sensitized Type 304 stainless steel will be examined and compared to determine how closely their corrosion behavior follows the above microstructural relationship. Determinations of the corrosion rates, anodic polarization behavior, and crevice environment in a simulated crack will complement the above work, and provide a basis for alternate corrosion test environments. Finally, the detailed understanding developed in this work will be extended in scoping studies on a variety of other stainless steels, such as 304LN, 316 and 347.

Organization: General Electric Co.

Principal Investigator: P. L. Andresen

Sponsor: EPRI (RP1072-1)

12. Corrosion Fatigue Characterization of Reactor Pressure Vessel Steels

Description: Present methods of evaluating crack growth rates under corrosion-fatigue conditions in reactor pressure vessels have proven to be inadequate. To prevent excessively conservative regulatory reaction in evaluating the safety of flaws found in future ISI, a corrosion fatigue materials data base will be developed and used to develop the predictive methodology for determining crack growth rates under actual reactor operating conditions.

Organization: Babcock and Wilcox (B&W), Westinghouse (W), NA

Principal Investigator: NA

Sponsor: EPRI [RP 1325-1 (B&W), -4 (W), -5 (NA)]

13. SCC of Stainless Steel and Inconels

Description: This program is an investigation of the problem of SCC in stainless steels and high-nickel alloys, metals used in the design and construction of nuclear reactors for electric power generation. The interaction of SCC and cyclic stressing will also be studied. Specific tasks include (a) defining conditions of environmental composition, alloy composition, and alloy structure under which stainless steels will be resistant to SCC; and (b) developing modified Inconel-type alloys that can be expected to perform reliably without SCC.

Organization: Ohio State University, Research Foundation

Principal Investigator: D. D. MacDonald

Sponsor: EPRI

14. The Influence of Combined Environmental Effects on the SCC of Welded Stainless Steel Piping

Description: The objective of this research is to determine the relative influences of oxygen concentration, hydrogen peroxide concentration, and temperature on IGSCC of welds in BWR piping. The purpose in developing this information is to provide guidance and technical justification for implementing oxygen and/or peroxide reduction methods within BWR plant operations.

Organization: Battelle-Columbus

Principal Investigator: W. Berry

Sponsor: NA

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